NUCLEAR SCIENCE ABSTRACTS

Volume 11 August 15, 1957 No. 15 Suppl.

BIOLOGY AND MEDICINE

8257 HW-41026(Del.)
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
RADIOLOGICAL SCIENCES DEPARTMENT RESEARCH AND DEVELOPMENT ACTIVITIES QUARTERLY PROGRESS REPORT [FOR] OCTOBER – DECEMBER 1955. H. M. Parker. Jan. 11, 1956. Decl. with deletions Feb. 26, 1957. 33p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported in the following studies: a radiobiological-ecological survey of the Columbia River; the toxic effects of continuous low-level exposure to I^{131} in pigs; Pu metabolism in miniature pigs and rats; the tissue distribution and retention of Ru^{106} and Cs in rats and mice; the radiosensitivity of the intestine of rats; the pulmonary absorption of radioactive particles and the pathological effects of such particles in the lungs of mice; the uptake of I^{131} and other fission products from the soil and from air by plants; determinations of the relative biological effectiveness for yeast cells of Po^{210} α particles and P^{32} β particles; the development of chemical and radiological monitoring methods for studies of radioactive contamination, temperature, and concentration of non-radioactive toxic materials in waste streams, soil, ground water, and air; the separation of fission products from waste streams; and improvements in instruments for monitoring and low-level radiation counting. The status of the research programs of the department is reviewed. (For preceding period see HW-39624.) (C.H.)

8258 UCLA-276
California. Univ., Los Angeles. Atomic Energy Project. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1953. Jan. 10, 1954. Decl. Mar. 6, 1957. 136p. Contract AT-04-1-GEN-12. $19.80(ph OTS); $6.30(mf OTS).

Results are reported of studies on x-ray effects on fat absorption in rats; nucleic acids in tobacco mosaic virus, leukocytes, and yeast; desoxyribonucleic acid molecules; synthesis of oleic acid-1-C^{14}; Borodin degradation of stearic acid; synthesis of 5,6,8,9-tetrahydroarachidonic acid; study of fatty acid degradation in 100 mg samples; chromatographic separation of fatty acids; evaluation of adrenergic drugs; x-ray effects on Fe in blood plasma; pulmonary absorption of particles; effects of injections on radiation effects; radiation effects on gastrointestinal function; sulfhydryl content of plasma and ovalbumin; soil particle size measurement; microtome modification for electron microscope histology; fission product uptake by plants from soil; metabolism of Sr by rats; production of pulmonary edema in cats by air blast; Au^{198} scintigrams of human liver; chemical dosimeters. (T.R.H.)

8259 UCLA-371
California. Univ., Los Angeles. Atomic Energy Project. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1956. July 1, 1956. Decl. Mar. 6, 1957. 112p. Contract AT-04-1-GEN-12. $16.80(ph OTS); $5.70(mf OTS).

Progress is reported in studies on the synthesis and analysis of unsaturated fatty acids; the effect of pre-treatment with intravenous iron on post-irradiation hyperferremia; the screening of various pharmacological agents for their effect on radiation injuries; the effect of particle and ciliary action size on pulmonary absorption and distribution of particulate matter; effects of irradiation on blood pressure in rats; effects of irradiation on the chemical composition of embryonic blood serum and isolated serum lipoproteins; the action of radiation on alkaline ferriprotoporphyrin solutions; the effect of weapon testing in the Pacific on background levels in the state of California; determination of the characteristics of the 60-in. cyclotron beam; the response of chemical radiation detectors to γ radiation and to neutrons; and the uptake and metabolism of fission products by plants. (For preceding period see UCLA-362.) (C.H.)

RADIATION HAZARDS AND PROTECTION

8260 HW-20055(Del.)
Hanford Works, Richland, Wash.
EFFECT OF HANFORD PILE EFFLUENT UPON AQUATIC INVERTEBRATES IN THE COLUMBIA RIVER. Jared J. Davis and Calvin L. Cooper. Jan. 19, 1951. Decl. with deletions Feb. 8, 1957. 63p. Contract W-31-109-Eng-52. $10.80(ph OTS); $3.90(mf OTS).

A radiological-ecological survey was made during the period of October, 1948 through February, 1950 of the invertebrate fauna that inhabit the Columbia River within the confines of the Hanford Works and downstream to the site of McNary Dam. Materials and methods are discussed and the results of extensive radioassays, qualitative and quantitative biological determinations, and hydrographic studies are given and analyzed. Radioactive wastes were not found to be causing any apparent deleterious effects on the natural invertebrate fauna. (C.H.)

TOXICOLOGY STUDIES

8261 HW-23332
Hanford Works, Richland, Wash.
RADIOLOGICAL SCIENCES DEPARTMENT RESEARCH AND DEVELOPMENT ACTIVITIES FOR OCTOBER–DECEMBER 1951. Quarterly Progress Report. Jan. 24, 1952. Decl. Feb. 20, 1957. 30p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported on research and development activities in biology and biophysics. Data are included from the radiobiological ecological survey of the Columbia River, a study of the effects of pile effluent water on aquatic organisms, a study of the absorption of Pu from the gastrointestinal tract, Pu toxicology studies, the effect of growth on the deposition of tissue-bound T, pulmonary absorption of T, effects of radiation on plant life and translocation of radioelements in plants, the retention of bound T oxide by the bean plant and T fixation by bacteria. The toxicology of I^{131} was investigated in sheep and some findings are included. (D.E.B.)

CHEMISTRY

8262 A-1246
Columbia Univ., New York. Div. of War Research.
AN INVESTIGATION OF THE PREPARATION OF BORON BY REACTION OF BORON COMPOUNDS WITH METALS. J. S. Spevack and A. N. Kurtz. May 25, 1944. Decl. Feb. 14, 1957. 56p. Contract W-7405-Eng-50. (100B-R-152). $15.30(ph OTS); $5.40(mf OTS).

Boron was prepared by reactions of active metals with boric anhydride and with boron–fluorine compounds. The preparation of crystalline boron by the reaction of BF_3 with Na appears to be commercially feasible. (C.W.H.)

8263 A-1298
Columbia Univ., New York. Div. of War Research.
RESEARCH ON METHODS OF PRODUCTION OF CRYSTALLINE BORON OF HIGH PURITY FROM BORON HALIDES. Clyde A. Hutchison, James S. Smith, and Jeremiah L. Thomas. Aug. 10, 1944. Decl. Feb. 12, 1957. 99p. Contract W-7405-Eng-50. (100B-R-183). $22.80(ph OTS); $7.20(mf OTS).

The reduction of BCl_3 under various experimental conditions (filaments, tubes, etc.) was investigated. Boron (~99 at. %) was obtained. Methods were devised for recovering BCl_3 from the exhaust gases and B from tantalum rods. (C.W.H.)

8264 ANL-4232(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-II SUMMARY REPORT FOR APRIL, MAY, AND JUNE 1948. Feb. 10, 1949. Decl. with deletions Feb. 12, 1957. 60p. Contract W-31-109-Eng-38. $9.30(ph OTS); $3.60(mf OTS).

Investigations are reported on: reactions of BeO with steam, atomic gases, and free radicals; properties of graphite; oxidation of graphite; distribution of C^{14} in irradiated graphite; radiation effects on MgO; alkali metal halide crystals with color centers; radioactivity of Be^{10}; effect of Na^+ on precipitation of U with NH_4OH; determination of Li in alundum and Al–Li; test for completeness of U separation from Fe by $(NH_4)CO_3$ method; and surface tension of liquid K. (T.R.H.)

8265 ANL-4740(Rev.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-1 SUMMARY REPORT FOR JULY, AUGUST, AND SEPTEMBER 1951. D. W. Osborne, ed. Nov. 1, 1951. Decl. Mar. 6, 1957. 35p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

The binding energy of four neutrons in U^{239} and the disintegration energies of U^{239} and Np^{239} are calculated and compared with experimental investigations of disintegration. The principal conclusion drawn from two volatility-program runs is that dissolution of U in BrF_3 in the presence of F_2 is capable of producing a still-feed containing no volatile Pu and with the fission-product activities in such a state that the UF_6 may be quantitatively separated. The general principle of separation of heavy elements by differential electrical migration, the selection of an electrolytic solvent, electromigration sequences, and cost of separation by electromigration are discussed. A table shows direction of migration of uranyl, zirconyl, Th, and cupric ions in 0.1M solutions of various acids. Spectra are shown for Ta^{5+} in oxalic acid reduced for 1 and 3 hr at 0.5 ma and 2.5 v. Graphs show experimentally determined magnetic susceptibilities of $NaNpO_2(C_2H_3O_2)_3$, $NpO_2C_2O_4H.2H_2O$, and $KNpF_6$ at various temperatures and compare these values with calculated values. The method used for measurement of the susceptibilities and the theoretical model on which the calculations are based are described. The melting curve of He^3 from 0.16 to 1.51°K is presented and the apparatus and techniques used in obtaining the data are described. (auth)

8266 ANL-4797(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-II SUMMARY REPORT FOR JULY THROUGH DECEMBER 1951. J. R. Gilbreath and O. C. Simpson. July 30, 1952. Decl. with deletions Feb. 11, 1957. 76p. Contract W-31-109-eng-38. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following investigations: physical properties of graphite; effect of pile irradiation on properties of graphite, diamond, SiC, B_4C, quartz, and SiO_2, x-ray-induced luminescence of ice; α and β bands in KBr crystals; absorption spectra of thin films of KI; long-lived phosphorescence of $KI-TlI_3$; mechanism of γ-ray-induced chain oxidation of aqueous $FeSO_4-HCO_2H-O_2$ solutions; γ-ray-induced decomposition of H_2O_2 in aqueous solutions; use of aqueous solutions for pile dosimetry; nuclear properties of Zr^{93} and Nb^{93m} from fission; angular correlation and intensity of inner bremsstrahlung from P^{32} and RaE; β spectra of Pu^{239}, Pu^{240}, Pu^{241}, Sm^{151}, and Tc^{99}; radiations of U^{237}; γ spectrum of Np^{237}; spectra of Ac; and determination of Li in Li–Al alloy. (J.S.R.)

8267 ANL-4861(Rev.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-I SUMMARY REPORT FOR OCTOBER, NOVEMBER AND DECEMBER 1951. D. W. Osborne, ed. Feb. 1, 1952. Decl. Mar. 6, 1957. 44p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30 (mf OTS).

Progress is reported on the following studies: half life of Pu^{241} (tentatively, 12.5 yr); neutron capture cross sections of Pu^{240} and Pu^{241}; fission and capture cross sections of higher Pu isotopes; the (n,2n) cross section of U^{238} and Th^{232}, energy cycles involving the β-decay energy of Th^{233}, U^{237}, and Pu^{241}; theory of the fission-to-capture cross section ratio; Np(IV)-Np(V) couple in perchloric acid; separation of U, Th, and Zr by electrical migration; the chloro complexes of Ru(IV); and modifications of the absorption spectra of anions induced by complex formation. (D.E.B.)

8268 BNL-201
Brookhaven National Lab., Upton, N. Y.
THE DISTRIBUTION OF ELEMENTS IN SALT–METAL SYSTEMS WITH SPECIAL REFERENCE TO THE DATA OF D. W. BAREIS. R. H. Wiswall, Jr. Sept. 30, 1952. Decl. Feb. 18, 1957. 14p. $0.25(OTS).

The behavior of solute elements in systems consisting of fused salt mixtures and molten bismuth supports the hypothesis that a process of chemical reaction rather than solvent extraction is involved. The nature and extent of these reactions are predictable from thermodynamic data. A wider application of such salt-metal systems in separation processes is foreseen. (auth)

8269 CC-1980
[Iowa State Coll., Ames.]
CHEMICAL RESEARCH: GENERAL FOR THE PERIOD SEPTEMBER 10, 1944 TO OCTOBER 10, 1944. F. H. Spedding and H. A. Wilhelm. Nov. 9, 1944. Decl. Feb. 16, 1957. 19p. (A-3015). $3.30(ph OTS); $2.40(mf OTS).

X-ray diffraction data are presented on the U–O system throughout the composition range UO_2 to U_3O_8. X-ray diffraction data are also presented on Th hydride prepared at 16 psi and 1800 psi. Data are presented on the dehydration of $CeCl_3$; the preparation of anhydrous ThF_4; the recovery of Th from slag and CaO liners from bomb reductions; the production of Ce by the reduction of anhydrous $CeCl_3$ by Ca; and the preparation of ThF_4 by the action of HF on $Th(C_2O_4)_2$. (C.H.)

8270 CC-3402
Chicago. Univ. Metallurgical Lab.
COLUMN ADSORPTION METHODS IN ANALYTICAL CHEMISTRY. Jack Schubert. Jan. 11, 1946. Decl. Feb. 15, 1957. 25p. Contract W-7401-eng-37. $4.80(ph OTS); $2.70(mf OTS).

Extensive investigations of ion exchange processes have been made as a result of the demand for radiochemical separation techniques. A discussion of the ion exchange process, characteristics of available ion exchangers, and a number of specific separation and concentration processes are given. (D.E.B.)

8271 CC-3576
[Chicago. Univ. Metallurgical Lab.]
THE DISSOLUTION OF THORIUM METAL AND THORIUM DIOXIDE IN HNO_3–HF AND HNO_3–$(NH_4)_2SiF_6$ MIXTURES. Work Done September 15, 1944–June 30, 1945. F. W. Schuler, F. L. Steahly, and R. W. Stoughton. Aug. 2, 1946. Decl. Feb. 15, 1957. 16p. Contract W-7405-eng-39. $3.30(ph OTS); $2.40(mf OTS).

A satisfactory method has been found for dissolving Th metal and ThO_2 in 25-12 heat-treated stainless steel vessels without excessive corrosion of the vessels. The dissolution medium consists of strong HNO_3 containing a small amount of fluoride or fluosilicate. The fluoride required has no harmful effects on a subsequent solvent extraction step for a separation of U isotopes from the dissolved Th. (auth)

8272 CF-51-7-110
Oak Ridge National Lab., Tenn.
EXTRACTION TEST IN A PULSE COLUMN USING A SLURRY FEED CONTAINING 20% SOLIDS. A. C. Jealous. July 24, 1951. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $0.25(OTS).

8273 CF-56-2-54
Oak Ridge National Lab., Tenn.
THORIUM OXIDE FOR REED REQUIREMENT. V. D. Allred, L. J. King, and E. M. Shank. Feb. 6, 1956. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A total of 6220 lb of high purity ThO_2 produced by the Chemical Technology Division was delivered to Y-12. This material represents the initial portion of the REED 10,000 lb requirement, and was prepared in accordance with A. S. Kitzes' specifications of May 4, 1955 or as subsequently revised by mutual agreement. The available chemical and physical properties for each drum of oxide shipped were submitted and the data compiled on the attached summary sheets. This oxide probably represents the purest material which can consistently be prepared in the present facility using available starting materials. (auth)

8274 CF-56-2-135
Oak Ridge National Lab., Tenn.
APPLICABILITY OF LUMICON AND OF X-RAY FLUOROSCOPE TECHNIQUE TO THE STUDY OF ThO_2 SLURRY PROPERTIES IN STAINLESS STEEL SYSTEMS. V. D. Allred and N. A. Krohn. Feb. 22, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The applicability of x-ray fluoroscopy with and without the use of an image intensifier as a means of investigating the physical properties of aqueous thorium oxide suspensions has been examined. It appears that distinct advantages over ordinary radiography can be gained by using high energy x-ray fluoroscopy, but no particular advantage will be gained by using an image intensifier at this time. It is not recommended that an image intensifier (Lumicon) be obtained for the present ThO_2 slurry program. There are distinct advantages to be gained by setting up a high energy (200 to 300 kv) x-ray fluoroscope however. A fluoroscope unit would not only be considerably cheaper, but also would be required in part for a setup using the Lumicon. Major disadvantage of the fluoroscope alone to one using an image intensifier such as the Lumicon would be the lack of mobility of the viewing equipment. (auth)

8275 CF-56-3-151
Oak Ridge National Lab., Tenn.
DIFFERENTIAL INFRARED ABSORPTION TECHNIQUES FOR CHARACTERIZING THE SURFACE OF ThO_2. V. D. Allred. Mar. 26, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The differential infrared absorption technique for studying the surface of solids has been investigated. This method was tested on samples of thorium oxide at the University of Utah and was found to have excellent sensitivity in the infrared region. It is recommended that the differential infrared absorption spectra be used as a tool in the characterization of solids in laboratory and engineering pumping studies. (auth)

8276 CF-56-4-120
Oak Ridge National Lab., Tenn.
FURTHER INFORMATION ON THE CAKE FORMED DURING SLURRY TEST 200A-3. D. G. Thomas. Apr. 30, 1956. 43p. Contract [W-7405-eng-26]. $7.80(ph OTS); $3.30(mf OTS).

The thorium-oxide cake formed during slurry test 200A-3 was from $\frac{1}{8}$ to $\frac{1}{4}$-inch thick, covered all surfaces of the circulating system and had a bulk density greater than 5.4 grams/cu. cm. A measurement of the bending stress at rupture gave a modulus of rupture of 360 lb/sq in. for a cake sample compared with a value of 2850 lb/sq in. for a similar specimen of Tennessee marble. Spectrographic and chemical analysis showed no large concentrations of contaminants or impurities, although the sulfate concentration was lower than in any other test where $Th(SO_4)_2$ had been added. Traces of aluminum and silicon were observed presumably originating from the firebrick in the calcining furnace. X-ray crystallite measurements showed no difference between the pumped slurry and the cake. Sedimentation particle size analysis showed a change in the standard deviation from the slurry to the cake in the direction of a broader particle size distribution in the cake. Although the data on the cakes do not include the 50 wt percent point, apparently the mean particle diameter of the cake from the pipe is significantly smaller than any of the samples taken from the circulating stream. A plausibility argument shows that a thin layer of cake was deposited shortly before the shutdown at 35 hours, the remainder of the cake being deposited within 36 hours after starting the loop up again. As yet there is no explanation of how or why the cake formed. (auth)

8277 CK-1040
Chicago. Univ. Metallurgical Lab.
METALLURGY OF 94; REPORT FOR MONTH ENDING NOVEMBER 6, 1943. Decl. Feb. 16, 1957. 7p. Contract [W-7401-eng-37]. (A-1426). $1.80(ph OTS); $1.80(mf OTS).

The vapor pressure of PuO_2 was determined between 1400 and 1800°C. Investigations were continued on the preparation of ceramic crucibles, reduction of UF_4 with Ca or Mg, and hot-wire deposition of U. (C.W.H.)

8278 HW-19096
Hanford Works, Richland, Wash.
THE REACTIONS OF HEXONE IN THE REDOX PROCESS. B. R. Jones and L. L. Burger. Sept. 1, 1950. Decl. Feb. 19, 1957. 22p. Contract W-31-109-eng-52. $3.30 (ph OTS); $2.40(mf OTS).

The initial reaction in a nitric acid–hexone system appears to be the relatively rapid combination of hexone with

small amounts of nitrous acid to form an intermediate which then reacts in secondary reactions to produce more nitrous acid, resulting in an autocatalytic decomposition of hexone. If the initial nitrous acid concentration is very low, the secondary reactions do not become significant at room temperature. The limiting nitric acid concentration for a sustained or autocatalytic reaction depends on the temperature. Reagents which produce nitrous acid accelerate the hexone decomposition while any mechanism for removing nitrous acid inhibits the reaction. (G.Y.)

8279 IDO-14155
American Cyanamid Co. Atomic Energy Div., Idaho Falls, Idaho.
COLLECTION OF DATA ON ALUMINUM NITRATE AND URANYL NITRATE SOLUTIONS. J. R. Kelley. Aug. 21, 1952. Decl. Feb. 27, 1957. 16p. Contract [AT(10-1)-177]. $3.30(ph OTS); $2.40(mf OTS).

Data are reported on the specific gravity versus Al $(NO_3)_3$ molarity and weight percent $Al(NO_3)_3$ for solutions containing HNO_3 and deficient in HNO_3; the density of aqueous $UO_2(NO_3)_2$ and $UO_2(NO_3)-6H_2O$ solutions versus weight per cent; and the refractory index measurements determined from molecular coefficients of $UO_2(NO_3)_2$ in water, hexone and varsol, $Al(NO_3)_3$ in water, $NaNO_3$ in water and HNO_3 in water, hexone, and versol. (F.S)

8280 KLX-1394(Del.)
Vitro Corp. of America, New York.
INDUSTRIAL APPLICATION OF GROSS FISSION PRODUCTS. Progress Report [Covering the period] January 1—July 31, 1953. Job 24. Aug. 17, 1953. Decl. with deletions Feb. 27, 1957. 44p. Contract AT(30-1)-850. $7.80(ph OTS); $3.30(mf OTS).

Recommendations are presented concerning the scope and direction which future research should take for the complete development of fission-product radiation sources for industrial use in food and pharmaceutical sterilization. Methods of concentrating fission products are reviewed, and a brief outline of each process and a discussion of its limitations are included. It was concluded that concentration of fission products by evaporation, and subsequent drying to a powder is the most feasible method for the preparation of fission-product wastes from the Purex separation process. Procedures are discussed in detail. Schematic sketches are included for a concentration plant and a packaging plant. Chemical and physical problems are discussed. Direct utilization of used Materials Testing Reactor fuel rods as radiation sources is discussed. It is pointed out that no chemical processing or source fabrication would be required prior to their use. Three concepts of a design of a pilot plant for utilizing MTR fission-product sources for industrial sterilization are presented. (C.H.)

8281 NYO-1320
Mallinckrodt Chemical Works, St. Louis.
THE SOLUBILITY OF CALCIUM SULFATE IN SOME RAFFINATES. O. J. Buckheim, R. M. Paine, and A. E. Ruehle. Mar. 9, 1951. Decl. Feb. 28, 1957. 8p. $0.25 (OTS).

The approximate solubility of $CaSO_4$ in several ether-saturated raffinates was measured as equilibrium was approached from the high side. The effect of varying concentrations of Fe, Ca, SO_4^{2-}, and HNO_3 on the solubility of $CaSO_4$ was determined. (auth)

8282 ORNL-286(Del.)
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1949. Sept. 14, 1949. Decl. with deletions Feb. 8, 1957. 212p. Contract W-7405-eng-26. $33.30(ph OTS); $9.60(mf OTS).

Using anion exchangers the following separations have been effected: Th from Pa and U; Pa from U; Cb, Ta and Pa; Zr from Cb; Hf from Zr. Investigation of the aqueous system $U^{4+}-U^{5+}-U^{6+}$ and the system $UO_3-UO_2SO_4-H_2O$ is continuing with studies of reaction rates, equilibrium constants, and diffusion current constants for the latter. The properties of the isotopes Xe^{135}, Cs^{135}, I^{129}, Mo^{93}, Mo^{99}, Sn^{113}, I^{134}, are studied. The absorption spectra of $PmCl_3$ and $Pm(NO_3)_3$ are reported measured. Studies in the nuclear chemistry of the rare earths are continued. Methods for preparation of $CHOH-C^{14}$ and $CH_3COOH-2-C^{14}$ are given. The solvent extraction of Hf from HCl and HNO_3 using TTA is studied along with silica gel absorption of Zr and Cb from Redox first cycle raffinate. Improvements in the Rala Process are being developed. In physical chemistry are studies of solvents fluid at −196°C and nuclear properties of I^{129} and I^{127}. Development of experimental apparatus, instruments and analytical procedures is reported. (T.R.H.)

8283 ORNL-795
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1950. Oct. 3, 1950. Decl. Mar. 4, 1957. 156p. Contract W-7405-eng-26. $24.30(ph OTS); $7.50(mf OTS).

Chemistry of Source, Fissionable, and Structural Elements. A much improved anion exchange column for the separation of Pa^{5+} and UO_2^{2+} has been divised using a mixture of HF and HCl as an eluent. A study of the hydrolytic behavior of Th^{4+} has been completed. A method for the isolation of millicurie quantities of Th^{234} from UF_6 feed-tank concentrates has been developed and is discussed. Phase studies of $UO_2SO_4-H_2O$ system from ice eutectic (−13°C) up to the critical point for H_2O (~377°C) have been completed. The aqueous UO_2F_2 solutions also were found to manifest abnormally small freezing point depressions and mean thermodynamic activity coefficients were computed. Nuclear Chemistry. Investigations on the Sn radionuclides have continued. Lens and scintillation spectrometer measurements on Nb^{96} have given a maximum β-energy of 0.75 ± 0.01 and a γ-ray energy of 0.85 ± 0.10 Mev. Further quantitative measurements on the previously known Kr^{85} have revealed a β-branching of 0.65 ± 0.15% in the decay of this isotope. Radio-Organic Chemistry. The following C^{14}-labeled compounds have been prepared; potassium acetate-2-C^{14}, acetic acid-2-C^{14}, sublimed malonic acid-2-C^{14}, and potassium cyanide-C^{14}. Chemistry of Separation Processes. Studies on the interaction of ThF_4 and phosphorus fluorides have continued. Further investigations in the TBP Process have been concerned with the factors governing the distribution of HNO_3 between aqueous and organic phases. Measurements of the electrodeposition of traces of Ru have shown that fractions unplatable on bright Pt electrodes may be deposited completely on Hg cathodes at sufficiently negative potentials. An enrichment of Li^7 up to 94.397 ± 0.021% from its natural abundance was observed by ion exchange using Dowex 50. Basic studies on the equilibrium and rate processes governing the performance of ion exchangers have continued. Chemical Physics. The group working on calorimetry of radioactivity has measured calorimetrically the energy absorbed in graphite. Radiation Chemistry. The decomposition of H_2O and aqueous solutions is being studied with the radiation from the Co γ-ray source. A spectroradiometer has been built for studying the spectral distribution of the light emitted from irradiated alkali halide crystals. Instrumentation. A highly precise and stable pulse generator has been built for testing high-grain wide-band amplifiers and differential discriminators. (W.L.H.)

8284 ORNL-1432
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1952. Mar. 4, 1953. Decl. Mar. 9, 1957. 73p. Contract W-7405-eng-26. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following studies: density of UO_2F_2 solutions; anion exchange in sulfate media; thermodynamics of complex ions; chemistry of fused salts; metal-metal halide systems; growth of RaE(Bi^{210})β spectrum in RaD(Pb^{210}); neutron capture cross sections of Pa^{233}; radiation characteristics of Sm^{145} and Pm^{145}; search for an isotope effect in Br addition to labeled methyl cinnamate; isotope effects in saponification of carboxyl-labeled esters at 25°C; isotope effect in dehydration of formic acid and hydrolysis of hydrocyanic acid; Wagner rearrangement studies; thermal rearrangement of amine nitrates; organic chemistry of solvents; ion exchange studies; r-f spectra; crystal structures of the modification of NH_4Br and NH_4I with the NaCl-type structure; nucleation of bubbles in superheated solutions by fission recoils; oxidation of $U(4^+)$ by γ rays; γ-ray effects in ice; γ irradiation of benzene solutions; reduction of $Ce(4^+)$ by γ rays; solubility of UO_3 in UO_2SO_4 and UO_2F_2 at 175 and 250°C; electric conductivity measurements of UO_2SO_4 and UO_2F_2; vapor pressures of aqueous UO_2SO_4 solutions; conductometric method for studying the kinetics of peroxide decomposition in $UO_2SO_4^-$, and alternate fuel media and possible corrosion inhibitors. (auth)

8285 ORNL-1717(Rev.)
Oak Ridge National Lab., Tenn.
ANALYTICAL CHEMISTRY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING APRIL 20, 1954. July 1, 1954. Decl. Mar. 2, 1957. 53p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

Progress is reported on the following studies: polarographic determination of U in the presence of Mo and in stainless steel and shales; spectrophotometric determination of Zr, Ti, and Cu in homogenous reactor solutions; ion-exchange separation and polarographic determination of Re in Mo ores and flue dusts and Eu in LiI; determination of Mn in separate valence states; reductimetric and coulometric determination of Fe in presence of U; determination of Al in aqueous solutions by potentiometric titration with $KBrO_3$; determination of Th and Zr in aqueous solutions with disodium dihydrogen ethylenediaminetetraacetate; analytical instrumentation; activation determination of I, Au, Th, Na, and U; radiochemical and spectrochemical analyses; and analysis of UO_2SO_4 solutions. (J.E.D.)

8286 ORNL-1940
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING JUNE 20, 1955. Oct. 27, 1955. Decl. Mar. 2, 1957. 94p. Contract W-7405-eng-26. $0.60(OTS).

The polymerization of Th^{4+} as a function of its hydrolysis in perchlorate and chloride media was studied. The corrosion inhibition of carbon and stainless steels by pertechnetate and CO, respectively, was observed. The potentials of the cell In, $In_2(SO_4)_3$(m), Ag_2SO_4(sat.), Ag were measured from 15 to 70°C with the $In_2(SO_4)_3$ concentration varying from 0.01 to 2.3 molal. Density, molar volume, and electrical conductance data are reported for the molten alkali halides. The self-diffusion coefficient of Na in molten $NaNO_3$ was determined. The production and identification of Tc^{97}, Tc^{98}, and Tc^{99} are described. Decay properties of Rh^{104}, Tc^{100}, Rb^{88}, Rb^{89}, and Ce^{137} are reported. The pinacol rearrangements of triphenylethylene glycol and 1,2-diphenyl-1-p-tolylethylene glycol, condensation of o-benzoyl-benzoic acid-carboxyl-C^{14}, and brominative decarboxylation of 3,5-dibromo-4-hydroxybenzoic acid were studied. An x-ray-diffractometer for liquids is described. Neutron-diffraction studies of $Ca(OH)_2$ are reported. Isopiestic measurements were made on several Dowex resins. Neutron irradiation of KH_2PO_4 produced P^{32} in orthophosphate, pyrophosphate, triphosphate, and two unidentified components. The behavior of Np in the Purex-metal recovery process was investigated. The reactor decomposition of $Th(NO_3)_4$ solutions containing UO_2SO_4, the effect of Tl^+ on γ-induced reduction of Ce^{4+}, γ-decomposition of H_2SO_4, the radiolysis of substituted butanes, and the polymerization of HCN and of HCN mixed with Kr, A, or He were studied. Investigations were initiated on the phase equilibria at high temperature of the following systems, $Na_2O-UO_3-CO_2-H_2O$, $MgO-UO_3-CO_2-H_2O$, $Li_2O-UO_3-CO_2-H_2O$, and $CuO-UO_3-SO_3-H_2O$. (C.W.H.)

8287 ORNL-2041
Oak Ridge National Lab., Tenn.
BULK ThO_2, A REACTOR MATERIAL. J. R. Johnson and J. M. Warde. May 24, 1956. Decl. Mar. 2, 1957. 9p. Contract W-7405-eng-26. $0.25(OTS).

ThO_2 can be extracted and processed to a very pure bulk metal for fabrication into solid fertile elements. No reaction occurs with water up to 300°C. No bond existed between ThO_2 and its Al jacket and there is no heat transfer. Experimental evidence indicates that the oxide is suitable for reactor use. (F.S.)

8288 ORNL-2171
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING JUNE 20, 1956. Oct. 4, 1956. Decl. Mar. 2, 1957. 10p. Contract W-7405-eng-26. $0.25 (OTS).

The processing of a special lot of Chalk River U solution by a modified Purex method led to the recovery of approximately 28 g of Np^{237}. Evidence was obtained for the occurrence of a valence shift of Np(V) to extractable Np(IV) in contact with TBP. The limits of acid strength both for extraction and partitioning were determined. The rate of production of Np by the n,2n reaction on U^{238} was found to be 0.3% of the Pu production rate. (auth)

8289 TID-5192
New Brunswick Lab., AEC, N. J.
NEW BRUNSWICK LABORATORY REPORT ON THE THORIUM PROGRAM FOR THE PERIOD ENDING FEBRUARY 15, 1952. Decl. Mar. 5, 1957. 16p. $3.30(ph OTS); $2.40(mf OTS).

ANALYTICAL PROCEDURES

8290 CF-56-4-31
Oak Ridge National Lab., Tenn.
PROCEDURE FOR THE DETERMINATION OF OXYGEN IN SODIUM AND NaK BY THE DISTILLATION METHOD. J. C. White. Apr. 5, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The working procedure for the determination of oxygen in sodium and NaK by the distillation method is given. The apparatus, which is based on that developed by the Argonne National Laboratory, can be connected to either dynamic or static systems and makes possible the direct sampling of alkali at operating temperatures. Sodium and NaK have been sampled with this device at temperatures as high as 1400°F. The alkali metal is transferred, under pressure, to the sampler where a fixed volume of the metal is retained in a removable hemispherical cup. The metal is then distilled from the oxide at a temperature of 800°F and at a

pressure of <5 microns of Hg. After the sampler has cooled, the cup is removed from the sampler and the oxide, which remains as a residue in the cup, is dissolved in water. The resulting solution of alkali metal hydroxide is then titrated to a methyl orange indicator end point with a standard solution of hydrochloric acid. The concentration of oxygen in the alkali metal is calculated on the basis of the equivalents of acid consumed in the titration, and the weight of the alkali metal, retained during sampling, in the cup. (auth)

8291 HW-13280
Hanford Works, Richland, Wash.
DETERMINATION OF ORGANIC ACIDS IN PROCESS SOLUTIONS. R. J. Brouns and C. W. Pollock. June 29, 1949. Decl. Feb. 27, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

A method was established for the estimation of volatile organic acids in aqueous process solutions containing UNH, nitric acid, ANN, sodium dichromate, and small amounts of hexone. The practice is to distill a 400 μl or 500 μl sample in the presence of an excess of phosphoric acid and ferrous sulfate under a high vacuum; a special apparatus utilizing a receiver cooled with a dry ice–isopropanol mixture is employed. The distillate is taken up in isopropanol and then titrated potentiometrically with standard potassium hydroxide solution. Since nitric acid and organic acids are present, two end points are observed. The potassium hydroxide added between these end points is equivalent to the organic acids. In the titration, CO_2 from the atmosphere or in the potassium hydroxide is a source of error. The former was avoided and correction was made for the latter. (auth)

8292 HW-14603
Hanford Works, Richland, Wash.
THE DETERMINATION OF FREE NITRIC ACID IN SOLUTIONS CONTAINING URANYL NITRATE, ALUMINUM NITRATE AND SODIUM DICHROMATE. Robert Lee Moore and H. R. Schmidt. Oct. 21, 1949. Decl. Feb. 19, 1957. 14p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40 (mf OTS).

A method is described for the determination of free HNO_3 in solutions containing $UO_2(NO_3)_2$, $Al(NO_3)_3$ and HNO_3. The U and Al are precipitated with KF and the acid titrated with standard alkali. A pH meter equipped with a glass electrode and a calomel reference electrode is employed to follow the titration. The hydroxide equivalent of the dichromate, if present, is subtracted from the total titre, however it is not necessary to remove the dichromate, as in the oxalate method. Analysis of a series of synthetic solutions gave an average error of 0.14 g/l HNO_3 or 0.88%. The method is applicable to hexone solutions and to hexone-saturated aqueous solutions and may be scaled down to a micro scale without serious loss of precision or accuracy. NH_4NO_3 and hydrazine are found to interfere due to evolution of NH_3. (auth)

8293 HW-18150
Hanford Works, Richland, Wash.
TERMINAL REPORT ON THE RADIOLANTHANUM LABORATORY. II. ANALYTICAL METHODS RESEARCH AND ADAPTATION. H. R. Schmidt and W. W. Marshall. June 1950. Decl. Feb. 19, 1957. 26p. Contract [W-31-109-eng-52]. $4.80(ph OTS); $2.70(mf OTS).

The terminal status of the Analytical Sections research on, and adaptation of, analytical methods destined for the control of the Rala Process using a one milliliter sample was described. All work has been covered either by brief reference to Hanford subject reports on completed phases of the work or by more detailed description of progress on uncompleted problems supplemented by reference to progress reports and research notebooks. (F.S.)

8294 K-1076
Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.
IMPROVEMENTS IN THE ASSAYING OF LITHIUM. A. E. Cameron. Oct. 22, 1953. Decl. Feb. 23, 1957. 15p. Contract W-7405-eng-26. $0.30(OTS).

The mass spectrometric assay of Li by slow electron bombardment of LiI vapor is concluded to be the most reliable method available. Only Cs interferes with the spectrum at mass 133 where it coincides with the Li^6I^+ ion, and the presence of such interference can be readily established. Conversion of a six inch radius, 60° deflection instrument for solid sample handling is described and operating procedures outlined. The reproducibility is indicated by results obtained on a "standard" sample over a period of some months. Effects caused by presence of hydrate are indicated as a possible source of non-agreement between laboratories, quite apart from variations in the natural abundance of the Li isotopes. (auth)

8295 K-1199
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
A RAPID AND PRECISE METHOD FOR DETERMINING URANIUM IN URANIUM HEXAFLUORIDE. C. A. Kienberger. July 6, 1955. Decl. Feb. 25, 1957. 15p. Contract W-7405-eng-26. $0.25(OTS).

A method is described for determining U in UF_6 that contains essentially no non-volatile impurities. Approximately 7 g of liquid UF_6 is transferred to a weighed fluorothene tube, weighed, frozen, and hydrolyzed in a 200 ml platinum dish. This uranyl fluoride solution is evaporated to dryness and the uranium is ignited in an atmosphere of superheated steam to urano-uranic oxide. The dish is weighed, and the percentage of uranium calculated from the weights of oxide and hexafluoride. An analysis by the new method is completed with 0.7 man hour of labor. Control samples of either pure UF_6, or hexafluoride containing several weight percent fluorocarbons are routinely analyzed along with production samples, with a precision of ± 0.08% of the value, and without significant bias. (auth)

8296 ORNL-1880(Rev.)
Oak Ridge National Lab., Tenn.
ANALYTICAL CHEMISTRY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING APRIL 20, 1955. H. P. Raaen, ed. May 6, 1955. Decl. Mar. 2, 1957. 56p. Contract W-7405-eng-26. $0.50(OTS).

Analytical procedures were investigated for the following determinations: Cr and Al in reactor solutions, Sn in Cu–Sn alloys, Co in Al–Co–Mn alloys, U in stainless steel $-UO_2$ plates, TTA in xylene, V in TiO_2, Al, Mg, and Ti in NiO_2, Am in Amex Process solutions, Cl_2, Br_2, and As in petroleum, and Ba in ThO_2. Other procedures that were investigated included determinations of Tc, Rh, $C_6H_5AsCl_2 \cdot 2H_2O$, Zr, Co, Hg, Sn, Cd, Ni, ethylenediamine–tetraacetic acid, Na, K, Li, Rb, Cs, Mg, Ca, and Ba. A method has been developed for the separation of radioiron from a fission-product mixture and the determination of its specific activity. The neutron activation cross section of Pt^{196} has been determined. A 4π coincidence method was developed for absolute beta counting. The half lives of Co^{56}, Fe^{59}, Se^{75}, Rb^{86}, In^{114}, Ba^{131}, and Pr^{143} were determined. The electrodeposition of Cu from uranyl sulfate solutions was investigated. Procedures for the separation of Th from fission products were described. Analytical instrumentation was related to the development of such instruments as the falling-drop densimeter, a velocity-servo automatic potentiometric titrator, and an automatic coulometric titrator. The corrosion of stainless steel in reactor solutions and slurries was studied. (C.W.H.)

8297 ORNL-1887
Oak Ridge National Lab., Tenn.
DETERMINATION OF CORROSION PRODUCTS AND ADDITIVES IN HOMOGENEOUS REACTOR FUEL. II. POLAROGRAPHIC DETERMINATION OF CHROMIUM. A. D. Horton and P. F. Thomason—H. P. Raaen, ed. Oct. 3, 1955. Decl. Mar. 2, 1957. 16p. Contract W-7405-eng-26. $0.30(OTS).

A satisfactory ion exchange–polarographic method was developed for the determination of either Cr^{6+} or total Cr in homogeneous reactor fuels. Total Cr is determined as Cr^{6+}, i.e., chromate, and in the same way as is Cr^{6+}, after Cr in the lower valence states is oxidized to chromate by $KMnO_4$. Chromate is separated from all interfering metal ions in the fuel by ion exchange on a Dowex-50 resin column. The chromate in the effluent is determined polarographically in approximately 0.75M NaOH as supporting electrolyte. A well-defined polarographic wave is obtained for the $Cr^{6+} \rightarrow Cr^{3+}$ reduction at a half-wave potential of -0.85v vs. the SCE. The relative standard deviation of the data for 2 μg of Cr^{6+} per ml was 2%; for 4 μg of total Cr per ml, it was 3%. An ion exchange–polarographic method was developed also for the determination of Cr^{3+}. Chromium(III) is separated from all interfering ions in the fuel by ion exchange on a Dowex 1 resin column. The Cr^{3+} in the effluent is determined polarographically in a 1M NH_3–1M NH_4Cl supporting electrolyte. The wave obtained at a half-wave potential of −1.42v vs. the SCE is poorly defined, and the method is not entirely satisfactory. (auth)

8298 Y-810
Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.
SPECTROGRAPHIC ANALYSIS OF GRAPHITE RAW MATERIAL FOR BORON AND VANADIUM. J. E. Paterson, H. D. Whitehead, and R. K. Bennett. May 18, 1950. Decl. Mar. 6, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Boron and V may be determined routinely to 0.1 and 4 ppm respectively in raw materials for graphite production and in finished graphite by the method described. The samples are prepared for analysis with a shear-plate pulverizer, and then are ashed with CaO to trap B and V. A spectrographic analysis is made on the CaO residue by a pyro-electric concentration method, and the resulting spectra are compared visually against standards prepared from V_2O_5 and H_3BO_4 ground into CaO. (auth)

CRYSTALLOGRAPHY AND CRYSTAL STRUCTURE

8299 A-1791
American Cyanamid Co. Stamford Research Labs., Stamford, Conn.
PRODUCTION OF 891 (CRYSTALLINE BORON): MICROSCOPICAL EXAMINATION OF CROSS-SECTIONS OF 891 AS DEPOSITED ON CORE. Progress Report. T. G. Rochow. Apr. 24, 1944. Decl. Feb. 12, 1957. 30p. Contract W-7401-eng-91. $6.30(ph OTS).

A microscopical examination of the cross sections of crystalline B as deposited on various materials was made. The photomicrographs which were made are shown. (B.J.H.)

8300 ORNL-1467
Oak Ridge National Lab., Tenn.
RECRYSTALLIZATION OF THORIUM. F. H. Eckert, E. J. Boyle, D. S. Eaton, R. J. Gray, and J. R. Erwin. Mar. 16, 1953. Decl. Mar. 6, 1957. 16p. Contract W-7405-eng-26. $0.30(OTS).

The recrystallization characteristics of Ames thorium and iodide thorium were determined. The effects of annealing temperature, annealing time, and cold work were followed by metallographic and hardness determinations. The results show that Ames thorium and iodide thorium recrystallize in the temperature range from 500 to 700°C. Iodide thorium recrystallizes at a slightly lower temperature than the Ames material. Increasing the carbon content to 0.13% has no significant effect on the recrystallization temperature of iodide thorium. The higher carbon-content material shows some increase in hardness after completion of recrystallization. This behavior is believed to be a result of precipitation of carbon at the higher annealing temperatures. (auth)

FLUORINE AND FLUORINE COMPOUNDS

8301 A-2365
[Standard Oil Co. of Indiana, Chicago.]
REACTIONS OF BORIC ACID AND BORIC OXIDE WITH METHYL ETHER–BORON FLUORIDE COMPLEX. A. N. Webb and W. L. Rittschof. Oct. 31, 1945. Decl. Feb. 12, 1957. 15p. Contract W-7418-eng-41. $3.30(ph OTS); $2.40(mf OTS).

Evidence was obtained for a chemical reaction between $B(OH)_3$ and the BF in methyl ether–boron fluoride complex, and for a similar reaction between B_2O_3 and the complex. In each case it appeared that a new compound had been formed which was postulated to be boron oxyfluoride with the empirical formula BOF. In the case of the $B(OH)_3$ reaction, definite proof of the empirical formula BOF for the reaction product has been obtained, and it is indicated that the compound probably exists as a trimer associated with one mol of methyl ether. In the case of B_2O_3, it was found that the rate of formation of $(BOF)_3$ was slow even at 90°C. Thus it is indicated that unreacted B_2O_3 may accumulate in the reboilers of the plant fractionation unit. An experiment was carried out in which the rate of reaction of B_2O_3 with complex was determined. (auth)

8302 A-3602
Carbide and Carbon Chemicals Corp. [K-25 Plant], Oak Ridge, Tenn.
EXPLOSIVE LIMITS OF MIXTURES OF FLUORINE (216) WITH FLUOROCARBONS, HYDROCARBONS, AND WATER VAPOR. J. K. Fox, D. S. Voss, and J. P. Brusie. Sept. 25, 1945. Decl. Feb. 12, 1957. 7p. $1.80 (ph OTS); $1.80(mf OTS).

The maximum concentration of F_2 in mixtures of fluorine, fluorocarbons, hydrocarbons, and water at 60°C that can be introduced without an explosion is ~13 to 15 vol. %. (C.W.H.)

8303 K-931
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
PHASE EQUILIBRIA OF THE TERNARY SYSTEM: URANIUM HEXAFLUORIDE–CHLORINE TRIFLUORIDE–HYDROGEN FLUORIDE. PART II. SOLID-LIQUID EQUILIBRIUM OF THE BINARY SYSTEM: CHLORINE TRIFLUORIDE–HYDROGEN FLUORIDE. W. S. Wendolkowski and E. J. Barber. July 31, 1952. Decl. Feb. 25, 1957. 11p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The solid-liquid phase equilibrium of the binary system: ClF_3-HF has been investigated over the composition range of 0 to 100 formula percent ClF_3 using the modified Skau warming and cooling curve techniques previously reported. The system is simple with a single binary eutectic at −88.8°C at a composition of 7.5 formula percent ClF_3. Large positive deviations from Raoult's law were observed;

but there was no evidence of a miscibility gap as solution theory might predict. The phenomenon of enantiotropy in ClF_3 was observed at −83.1°C. (auth)

8304 K-1279
Oak Ridge Gaseous Diffusion Plant, Tenn.
MEASUREMENT OF FLUORINE CONCENTRATION IN GAS MIXTURES WITH THE ACOUSTIC GAS ANALYZER. B. J. Bogardus and R. C. Smith. June 15, 1956. Decl. Feb. 25, 1957. 34p. Contract W-7405-eng-26. $0.40 (OTS).

An acoustic gas analyzer has been adapted for the determination of fluorine in plant gases. Prior to analysis, bromine was chemically substituted for Fe in the gas mixtures and the UF_6 and HF were removed by cold and chemical traps. Calibration data are included. (C.W.H.)

GRAPHITE

8305 ANL-4599
Argonne National Lab., Lemont, Ill.
NOTES ON THE MAGNETIC SUSCEPTIBILITY OF NEUTRON-IRRADIATED GRAPHITE. Mark Goldsmith. Apr. 17, 1951. Decl. Feb. 13, 1957. 17p. Contract W-31-109-eng-38. $0.30(OTS).

The magnetic susceptibility of samples of graphite was measured before and after irradiation. The susceptibilities were remeasured after annealing in vacuo at elevated temperatures. An attempt was made to relate magnetic susceptibility data to estimates of the number of displaced C atoms induced by fast neutron bombardment. (C.H.)

8306 BMI-1010
Battelle Memorial Inst., Columbus, Ohio.
ANNEALING OF CRYSTAL DISTORTION IN IRRADIATED GRAPHITE. Alfred E. Austin and Ralph J. Harrison. June 20, 1955. Decl. Feb. 13, 1957. 50p. Contract W-7405-Eng-92. $0.45(OTS).

Crystal-structure distortion has been studied for moderator graphite irradiated at Hanford from 400 to 1500 Mwd/ct. The degree of recovery of crystal structure has been observed after annealing from 200 to 2500°C. This work has shown that there is appreciable contraction of irradiated graphite within layer planes along the a axis, in addition to the normally observed expansion in the c axis. This contraction, 0.8 per cent per 1000 Mwd/ct, can affect appreciably the average physical expansion. Marked recovery of intralayer distortion occurs with annealing at 200 C for 1 hr, while there is only slight change in interlayer distortion. The nature of the intralayer distortions was studied by Fourier analysis of X-ray diffraction line contour, using the (110) reflection. Interpretation of the derived Fourier coefficients indicated that the intralayer distortion is principally strain and there is relatively little crystal breakup for irradiation up to 1000 Mwd/ct. Above this irradiation level there is some crystal breakup. The elastic energy corresponding to the intralayer strains has been estimated. Energies of about 40 and 107 cal/cm^3 were obtained for specimens with 400 and 1000 Mwd/ct exposure. The 200 C anneal relieved two-thirds to three-quarters of these energies. (auth)

8307 CC-1830
Chicago. Univ. Metallurgical Lab.
THE WIGNER EFFECT IN GRAPHITE. RELATIONSHIP BETWEEN ADVANTAGE FACTOR AND LOCATION OF SAMPLES IN AN OPERATING PILE. Problem Assignment No. 323-MLC-2301. A. Novick, T. J. Neubert, and M. Burton. Oct. 20, 1944. Decl. Feb. 16, 1957. 23p. $4.80(ph OTS); $2.70(mf OTS).

The extent of the Wigner effect in graphite at any point in a pile is sensitive to the local structural situation. It is greater near a slug. These control experiments give information on changes in graphite to be expected in selected places in the pile. (F.S.)

8308 CC-3643
Argonne National Lab., Lemont, Ill.
RADIOCARBON FROM PILE GRAPHITE; CHEMICAL METHODS FOR ITS CONCENTRATIONS. James R. Arnold and W. F. Libby. Oct. 10, 1946. Decl. Feb. 16, 1957. 27p. Contract W-31-109-eng-38. $4.80(ph OTS); $2.70 (mf OTS).

Samples of pile graphite, irradiated in a test-hole at Hanford for 15 months, have been assayed for C^{14}, yielding 0.38 ± 0.04 microcuries per gram. At this level of activity, the pile graphite contains very valuable amounts of C^{14}. The relation between the above assay and the probable average assay of pile graphite is discussed, and it is concluded that the latter is almost certainly above 0.3 μc/gram. Controlled oxidation of this graphite, either with oxygen at ~750°C, or with chromic acid "cleaning solution" at room temperature, yields early fractions which are highly enriched in C^{14}. Concentrations of 5-fold with oxygen, and 50-fold with CrO_3, have been observed. The relation between the observed enrichment and the Wigner effect is discussed, and a mechanism accounting for the observations put forward. According to this, about 25% of the stable carbon atoms in the lattice have been displaced by Wigner effect, a large fraction of which have healed by migrating to crystal edges. All the C^{14} atoms have been displaced, and the same fraction of these migrate to the edges. The enrichment then results from surface oxidation, in the oxygen case. Predictions are made on the basis of this hypothesis. A technique of counting radioactive CO_2 in the gas phase is described. (auth)

8309 CP-2890
[Carnegie Inst. of Tech., Pittsburgh.]
THE HALL EFFECT IN NEUTRON IRRADIATED AND ANNEALED GRAPHITE. Robert J. Maurer and Richard C. Ruder. Apr. 25, 1945. Decl. Feb. 14, 1957. 8p. Contract W-7405-eng-277. $1.80(ph OTS); $1.80(mf OTS).

Irradiated Whiting graphite possesses a small negative or a positive Hall constant, depending upon the dosage. Annealing at 500°C for 1 hour in He results in a Hall constant which is large in magnitude and negative in sign. The properties of the annealed graphite are presumably close to those of the original unirradiated material. Unirradiated AGOT-K graphite has a large, negative Hall constant. A mechanism is suggested which accounts for the change in sign of the Hall constant as a result of neutron bombardment. (auth)

8310 NAA-SR-160
North American Aviation, Inc., Downey, Calif.
A 'BOND' THEORY FOR c-AXIS EXPANSION AND STORED ENERGY OF RADIATION DAMAGED GRAPHITE. G. J. Dienes. Nov. 27, 1951. Decl. Feb. 27, 1957. 36p. Contract AT-11-1-Gen-8. $6.30(ph OTS); $3.00(mf OTS).

An ab initio calculation of c-axis expansion, stored energy, and activation energy for annealing of radiation-damaged graphite is described. The theory is based on known properties of carbon-carbon bonds and of the interplanar interactions in graphite. The theory is restricted to single crystals and to temperatures where interstitial atoms and vacancies are the important lattice disturbances. c-axis expansion is treated as an equilibrium between two opposing forces: the repulsive forces associated with the interstitials which tend to force the planes apart, and the cohesive forces between the graphite planes. The stored energy of the lattice is evaluated as the sum of the energies required to form a vacancy and to place an interstitial

between the planes. The latter energy decreases quite markedly as the lattice expands. The activation energy for motion of vacancies and interstitial atoms has been evaluated. The interstitial atoms are easier to move around in an expanded lattice; and consequently, the activation energy for moving interstitals is a sensitive function of the c-axis expansion and hence of the concentration of interstitials. The kinetics of annealing with variable activation energies is discussed and the implications to the theory of annealing in radiation damaged graphite pointed out. Experiments are suggested to test the theory and to gain further insight into the mechanism of radiation damage and its annealing. (auth)

8311 NAA-SR-222
North American Aviation, Inc., Downey, Calif.
EXPERIMENTS ON GRAPHITES OF HIGH DENSITY AND LOW PERMEABILITY. G. A. Bennett and H. Pearlman. Feb. 18, 1953. Decl. Mar. 2, 1957. 46p. Contract AT-11-1-GEN-8. $0.40(OTS).

Exploratory experiments were carried out on commercially available graphites of high density and of low permeability. Also, methods have been developed for improving these properties in regular production grades of graphite. The following measurements were made: density, pore-size distribution, permeability toward air and liquid metals, and chemical analysis. Attempts were made to prepare impermeable graphite, starting with National Carbon Company type AUF graphite. The procedure was to deposit carbon in the pores either by thermal decomposition of a gas, such as CH_4 or C_2H_2, or by pyrolysis (at temperatures up to 2000°C) of furfural which had been impregnated in the graphite and polymerized by mineral acids. Reduction to a few per cent of the original air permeability measured at room temperature was readily obtained. The reduced permeability, particularly, in the case of gas decomposition, appeared to be associated principally with a thin layer near the outer surface of the sample. Short-term static tests of samples treated by either procedure showed no penetration by liquid Bi at 625°C under 200 psig. Dynamic tests showed that samples were unaffected after 8 hr exposure to molten Pb at about 300°C under 120 psig. Untreated samples leaked badly at less than 30 psig in the latter test. (auth)

8312 NAA-SR-247
North American Aviation, Inc., Downey, Calif.
HIGH TEMPERATURE DIFFUSION OF INDIVIDUAL FISSION PRODUCTS FROM SMALL URANIUM-IMPREGNATED GRAPHITE SAMPLES UNDER DEUTERON BOMBARDMENT. C. T. Young. May 28, 1953. Decl. Mar. 4, 1957. 45p. Contract AT-11-1-GEN-8. $0.40 (OTS).

An investigation was made of the diffusion of a number of major fission elements from small uranium-impregnated graphite plates during the actual creation of these fission elements and their precursors. This was accomplished by heating the graphite for about 20 hr in the range 1050° to 1950°C, while bombarding it with deuterons on the 60-in. cyclotron at the Crocker Radiation Laboratory at Berkeley. The apparent diffusions of several of the end members of decay chains were found to be influenced by diffusion of early members of the chain. Deuteron fission yields with an accuracy of ±30% were obtained for the fission elements for each of the bombardments as a check for recovery of material. (auth)

8313 NAA-SR-Memo-129
North American Aviation, Inc., Downey, Calif.
NAA GRAPHITE SUMMARY TO NOVEMBER 1, 1951. R. L. Carter. Nov. 26, 1951. Decl. Feb. 27, 1957. 13p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

8314 ORNL-307
[Oak Ridge National Lab., Tenn.]
NEUTRON DAMAGE IN GRAPHITE. Hubert M. James. Mar. 22, 1949. Decl. Mar. 12, 1957. 24p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Estimates are given of (a) the energy stored in graphite crystals and in displaced C atoms, per neutron of a given initial energy, and (b) the relative damage produced by the neutron distributions existing at several points in a pile. (L.M.T.)

LABORATORIES AND EQUIPMENT

8315 CF-56-2-48
Oak Ridge National Lab., Tenn.
COMPONENT STUDY FOR A 36-IN. RIGGINS CONTACTOR: PROBLEM STATEMENT. C. W. Hancher. Feb. 10, 1956. Decl. Mar. 16, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An outline is proposed for the development of a resin moving hydraulic system and component parts for a 36- or 48-in. diameter Higgins-type, semi-continuous, ion exchange contactor. (B.J.H.)

8316 HW-26459
Hanford Works, Richland, Wash.
THE VALVE-ACTUATED PULSE COLUMN—II. A STUDY OF THE TEMPERATURE EFFECT AND DESIGN VARIABLES. Leland L. Burger and L. H. Clark. Feb. 16, 1953. Decl. Feb. 28, 1957. 23p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

The valve-actuated pulse column is employed to give phase separation on each half cycle in the countercurrent flow cycle and provides independent control over each step in the cycle. The dependence on temperature of operation, on pulse frequency and amplitude, and on plate design and spacing has been studied for extraction and stripping of U with TBP in a hydrocarbon diluent. It is found that the efficiency increases quite significantly as the temperature is raised regardless of the direction of transfer and this increased efficiency is accompanied by greater operational freedom. A decrease in hole size, decrease in plate separation and increase in frequency all lead to higher efficiency in agreement with the concept that small drops continually formed coalesced and reformed, lead to maximum interphase transfer. The high extraction efficiency of the column is probably aided further by the relatively sharp pulse produced. (auth)

8317 LWS-24620
California Research and Development Co., Livermore, Calif.
PROPOSED NEW DESIGN OF LIQUID–LIQUID CONTACTOR. H. Schneider. Mar. 25, 1952. Decl. Feb. 27, 1957. 13p. (CRD-T4C-111). $3.30(ph OTS); $2.40(mf OTS).

A schematic drawing is presented of a liquid-liquid contactor for separation processes involving radioactive materials. Sketches of several proposed modified versions are included. The unit is essentially a horizontal pulsed extractor having a system of baffles and perforated plates to provide dispersion of each phase in the other. A working model was constructed and put into operation. Details of the model and performance data are included. (C.H.)

RADIATION CHEMISTRY

8318 CF-56-5-41
Oak Ridge National Lab., Tenn.
REMOVAL OF RADIOISOTOPES FROM CONTAMINATED

WATER BY ORGANIC COMPLEXING AND ADSORPTION OF ACTIVATED CHARCOAL. William J. Lacy and Don C. Lindsten. May 9, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Procedure, principles, and data for the removal of 13 radioisotopes from water are given. (M.H.R.)

RADIATION EFFECTS

8319 ANL-4604(Rev.)
Argonne National Lab., Lemont, Ill.
REDUCTION OF IRRADIATION DAMAGE TO URANIUM BY SMALL ZIRCONIUM ADDITIONS. Interim Report. Samuel Untermyer. Mar. 27, 1951. Decl. Feb. 13, 1957. 7p. Contract W-31-109-Eng-38. $0.25(OTS).

8320 BNL-1811
Brookhaven National Lab., Upton, N. Y.
EFFECTS OF GAMMA IRRADIATION ON NITRO ALKANE FORMATION IN THE PUREX AND REDOX SYSTEMS. P. Colombo. Apr. 20, 1954. Decl. Feb. 14, 1957. 4p. Contract AT-30-2-Gen-16. $1.80(ph OTS); $1.80(mf OTS).

Data are included from analyses of system samples indicating that no appreciable formation of nitro-paraffins occurs during γ irradiation of Purex and Redox systems. (T.W.S.)

8321 NAA-SR-1033
North American Aviation, Inc., Downey, Calif.
THE CHEMICAL EFFECTS OF 1 Mev ELECTRONS ON BrF_3 AT 25°C. S. J. Yosim. Oct. 1, 1954. Decl. Mar. 2, 1957. 44p. Contract AT-11-1-GEN-8. $7.80(ph OTS); $3.30(mf OTS).

An investigation of the chemical effects of 1-Mev electrons on BrF_3 at 25°C has been carried out. Pressure measurements taken during the irradiation suggested the presence of Br_2 and BrF_5 as decomposition products and a fractional distillation of the irradiated liquid confirmed their presence. The extent of decomposition was determined both by fractional distillation and spectrophotometric methods. The radiation effect seemed to reach saturation when approximately 10 per cent of the BrF_3 was destroyed. The exposure necessary for the decomposition products to reach a concentration of half the saturated value was calculated to be 2.7 microampere hours/cc BrF_3, while the "G" value was found to be 1.5. A qualitative comparison of irradiation dosages from the Statitron with that expected from spent fuels revealed that little decomposition of BrF_3 reagent is to be expected from 1-day cooled Hanford fuel while in the case of 1-day cooled MTR type fuel (in pile for 12 days) a saturated effect might be realized in 1 to 3 hours. Since at most only 10% of the BrF_3 is destroyed it is concluded that BrF_3, from a radiation resistance standpoint, is a suitable reagent for the processing of short cooled fuels. (auth)

RARE EARTHS AND RARE-EARTH COMPOUNDS

Refer to abstract 8404.

SEPARATION PROCEDURES

8322 ANL-4228
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT FOR JULY, AUGUST AND SEPTEMBER 1948. Decl. Feb. 14, 1957. 41p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30(mf OTS).

First cycle Redox column studies show that the hexone solvent can be recovered by a caustic wash followed by distillation. A special grade of hexone is available which needs no acid-dichromate pretreatment and may be reused several times before washing and distillation are necessary. A proposal to decontaminate without U–Pu separation in the first cycle followed by second cycle decontamination and separation and third cycle decontamination was investigated. Studies into the removal of Ru by oxidation of the sulfate with O_3–O_2 followed by volatilization of the RuO_4 are reported. (T.R.H.)

8323 ANL-4530
Argonne National Lab., Lemont, Ill.
TRIBUTYL PHOSPHATE SOLVENT EXTRACTION PROCESS FOR RECOVERY AND DECONTAMINATION OF URANIUM 235 FUEL DISCHARGED FROM EXPERIMENTAL BREEDER REACTOR. Leslie Burris, Jr. and Richland C. Vogel. Sept. 15, 1950. Decl. Feb. 13, 1957. 144p. Contract W-31-109-eng-38. $22.80(ph OTS); $7.20(mf OTS).

Laboratory and semi-works studies on the development of a solvent-extraction separations process for enriched uranium are described. The process, using TBP diluted with CCl_4 as the solvent, was developed to meet the decontamination and recovery requirements of enriched fuel discharged from the Experimental Breeder Reactor. 37 figures. (C.H.)

8324 ANL-5109(Del.)
Argonne National Lab., Lemont, Ill.
AN IMPROVED CONTINUOUS ETHER EXTRACTOR FOR THE DETERMINATION OF URANIUM IN DISSOLVER SOLUTIONS. R. W. Bane and K. J. Jensen. Aug. 27, 1953. Decl. with deletions Feb. 11, 1957. 6p. Contract W-31-109-Eng-38. $1.80(ph OTS); $1.80(mf OTS).

An improved continuous ether extractor is described. The modifications include a means of safely disposing of the active raffinate and a means of positively checking the raffinate for completeness of uranium extraction. The results obtained on synthetic samples and dissolver solutions are given. (auth)

8325 BMI-JDS-151
[Battelle Memorial Inst., Columbus, Ohio]
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. A. E. Bearse, E. G. Bohlmann, G. D. Calkins, and R. B. Filbert, Jr. Nov. 15, 1948. Decl. Feb. 15, 1957. 56p. Contract AT-30-1-gen-228. $0.50(OTS).

Results on the NaOH–HNO_3 process for dissolving monazite sand and providing a solution for the recovery of Th and U by solvent extraction methods continue to show promise. Progress has been made in working out the conditions necessary to obtain a nitric acid solution containing a high percentage of the Th and U initially present in the sand and containing only small amounts of phosphate. (auth)

8326 BMI-JDS-188
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Quarterly Report [for] February 15, 1949 to May 15, 1949. A. E. Bearse, E. G. Bohlmann, G. D. Calkins, and R. B. Filbert, Jr. Decl. Feb. 15, 1957. 63p. Contract AT-30-1-Gen-228. $10.80(ph OTS); $3.90(mf OTS).

Chemical, ion exchange, and precipitation separation from ore, solvent extraction of U from Th, and chemical and physical separation of Ti are reported. (D.E.B.)

8327 BMI-JDS-209
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Quarterly Report for May 15, 1949 to August 15, 1949. G. D. Calkins, R. B. Filbert, Jr., R. H. Poirier, and M. A. Youtz. Aug. 15, 1949. Decl. Feb. 12,

1957. 89p. Contract AT 30-1-gen-228. $13.80(ph OTS); $4.80(mf OTS).

8328 BMI-JDS-219
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Progress Report for the Month of November 1949. G. D. Calkins, R. F. Devine, R. B. Filbert, Jr., S. J. Kiehl, Jr., and R. H. Poirier. Nov. 30, 1949. Decl. Feb. 12, 1957. 21p. Contract AT-30-1-gen-228. $3.30(ph OTS); $2.40(mf OTS).

Investigations are presented for the separation of Th and U from monazite sands, dissolution of monazite and treatment with NaOH followed by mineral acid, dissolution of Th and U from hydroxide precipitates by carbonate leaching, and purification by solvent extraction. (W.L.H.)

8329 BMI-JDS-216
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Progress Report for the Month of October 1949. G. D. Calkins, R. B. Filbert, Jr., and R. H. Poirier. Oct. 31, 1949. Decl. Feb. 12, 1957. 22p. Contract AT-30-1-Gen-228. $4.80(ph OTS); $2.70(mf OTS).

8330 BMI-JDS-217
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Quarterly Report for August 15 to November 15, 1949. G. D. Calkins, R. A. Ewing, R. B. Filbert, Jr., R. H. Poirier, S. J. Kiehl, Jr., and M. A. Youtz. Nov. 15, 1949. Decl. Feb. 12, 1957. 84p. Contract AT-30-1-Gen-228. $13.80(ph OTS); $4.80(mf OTS).

8331 BNL-125
Brookhaven National Lab., Upton, N. Y.
A CONTINUOUS FISSION PRODUCT SEPARATION PROCESS. I. REMOVAL OF THE RARE EARTHS (LANTHANUM, CERIUM, PRASEODYMIUM, AND NEODYMIUM) FROM A TYPICAL LIQUID BISMUTH–URANIUM REACTOR FUEL BY CONTACT WITH FUSED LiCl–KCl MIXTURES. D. W. Bareis. July 1, 1951. Decl. Feb. 14, 1957. 23p. Contract AT-30-2-gen-16. (LFR-3). $0.25 (OTS).

An investigation of the distribution of the rare earth elements between a liquid Bi–U solution and fused KCl–LiCl mixtures indicated that a separation process making use of a two-phase extraction involving the above system would be both practical and inexpensive. Further research on the liquid Bi–U-fused salt system is warranted in view of the benefit to be derived from a continuous fission product removal process for making the thermal breeder reactor a reality. (auth)

8332 CF-50-4-1
Oak Ridge National Lab., Tenn.
ORNL 25 RECOVERY PROCESS DEVELOPMENT; FISSION BURNUP OF ENRICHED URANIUM USED IN PROCESS VERIFICATION. R. B. Briggs and A. M. Rom. Apr. 28, 1950. Decl. Feb. 15, 1957. 15p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Enriched (93.5%)U^{235}–Al alloy was used in the final pilot plant verification of the 25 Recovery Process developed at ORNL for the reclamation of unspent U in fuel assemblies discharged from the Materials Testing Reactor. The amount of burnup had been calculated prior to pilot plant processing of the enriched alloy, but additional data acquired after processing made possible a more accurate calculation of U burnup on the basis of isotopic abundances. A calculation based on pile operating data is also included. (auth)

8333 CF-51-1-10
Oak Ridge National Lab., Tenn.
A REVIEW OF THE RELATIVE MERITS OF PACKED COLUMNS, PULSE COLUMNS, MIXER-SETTLER MECHANISMS, AND CENTRIFUGAL CONTACTORS. A. C. Jealous and E. C. Stewart. January 8, 1951. Decl. Feb. 13, 1957. 11p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Extensive experimental, pilot plant, and size scale-up studies on packed columns have established the reliability of this type contactor for project applications. Pulse columns and mixer-settlers are presently undergoing intensive project development, because they appear to present certain advantages over the packed column contactor. A clear-cut choice of type of contactor best suited to particular project applications is not presently possible. However, the comparisons offered in the attached tabulation with respect to the Purex Process may serve a useful purpose for guidance of further contactor studies or for an immediate or preliminary choice of contactor in the event that this is necessary. Currently planned contactor development will permit a closer analysis and, hence, a sound selection for specific process application at a later date. (auth)

8334 CF-51-6-15
Oak Ridge National Lab., Tenn.
PLUTONIUM ISOLATION FLOWSHEET. E. B. Sheldon. June 6, 1951. Decl. Feb. 20, 1957. 7p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Flowsheets are presented for the plutonium isolation from Purex IIBP streams by ion exchange. (C.W.H.)

8335 CF-51-6-149
[Oak Ridge National Lab., Tenn.]
ADDITIONAL PULSE COLUMN TESTS FOR URANIUM ORES RECOVERY PROCESS. A. C. Jealous. June 29, 1951. Decl. Feb. 20, 1957. 8p. Contract [W-7405-eng-26]. $0.25(OTS).

Scrub pulse column test runs for the Slurex Process are reported. Pulsing conditions of a 0.7-in. stroke, 50 strokes/min at 20 and 50°C for the 25-ft column were satisfactory in that no flooding was observed. (T.R.H.)

8336 CF-51-7-192(Del.)
[Oak Ridge National Lab., Tenn.]
CONCENTRATION OF TBP PROCESS RAFFINATE AND RECOVERY OF HNO_3 CONTENT. T. A. Arehart. July 2, 1951. Decl. with deletions Feb. 14, 1957. 24p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Recovery of 70 to 90% of the acid content of "hot" aqueous raffinate from TBP metal recovery process, with adequate decontamination of condensate and recovered acid was realized in a small shielded pilot plant evaporator. (auth)

8337 CF-51-10-204(Del.)
Oak Ridge National Lab., Tenn.
DESIGN SPECIFICATION AND CALCULATIONS FOR MTR DISSOLVER CONDENSER. W. L. Carter and J. M. Holmes –L. H. Landrum, comp. Oct. 31, 1951. Decl. with deletions Feb. 14, 1957. 59p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

8338 CF-51-11-56
Oak Ridge National Lab., Tenn.
ION EXCHANGE STUDIES. Quarterly Report for the Period August 10–November 10, 1951. I. R. Higgins. Nov. 7, 1951. Decl. Feb. 15, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

The application of ion exchange to processing of highly

radioactive solutions is frequently questioned because of the effect of the radiation on the organic adsorber. From previous studies using a Co^{60} γ source, Dowex 50 resin lost 10 to 15% of its capacity per whr/g of resin. This was checked by observing the damage from adsorbed β radiation. The β energy was calculated to be 0.00154 w/c and 100% adsorption of the β was assumed. On three checks by titration on the basis of oven dry resin the capacity loss was 9.5, 15.9, and 20.7% whr/g. The mechanism of the damage is believed to be a hydrolysis of the sulfonate groups on the resin to give free SO_4. Enough free SO_4 was found to account for 11.4% capacity loss on the sample that titrated 9.5%. Three independent free SO_4 checks indicated 60, 48, and 50 Meq of SO_4 released per whr. The Co^{60} γ tests indicated that the phenolic cation exchangers lost capacity by a factor of 2 or 3 slower than the styrene resins. (auth)

8339 CF-52-1-194
Oak Ridge National Lab., Tenn.
FINAL SLUREX RUNS. A. C. Jealous. Jan. 30, 1952. Decl. Feb. 15, 1957. 6p. Contract W-7405-eng-26. $0.25(OTS).

The slurex extraction process with the organic phase continuous yielded slightly higher U recovery and appeared to eliminate the problem of carry-over of insoluble waste in the organic product. In the course of stripping the U from the organic phase it was found that uncoated stainless steel plates had three times the capacity of Teflon-coated stainless steel plates with the same stripping efficiency. (D.E.B.)

8340 CF-52-2-149
Oak Ridge National Lab., Tenn.
SOLVENT EXTRACTION: INTERIM "23" PROCESS—QUARTERLY REPORT FOR PERIOD NOVEMBER 10, 1951 TO FEBRUARY 10, 1952. J. E. Conaway and A. C. Jealous. Feb. 18, 1952. Decl. Feb. 15, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A separation column consisting of 20 ft each of scrub and extraction sections and a strip column height of 24 ft were recommended. On the basis of a normal processing rate of 100 kg Th/day and a maximum of ~140 kg Th/day 2-in. I.D. was recommended for both columns. Test runs at the 100 kg Th/day rate yielded a Th separation factor of 10^5 and 0.1 to 0.2% U loss through each column. These data are based on 4/7.5/2/1.25 flow ratios which were adopted when it was found that the original 4/5/1/1.25 ratios resulted in excessive column lengths. (auth)

8341 CF-52-2-150
Oak Ridge National Lab., Tenn.
ISOLATION OF U-233 FROM IRRADIATED THORIUM USING DILUTE TBP AS THE EXTRACTANT. Quarterly Report. C. V. Ellison. Feb. 19, 1952. Decl. Feb. 15, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

One hot run using a Th metal slug irradiated 7 days in the LITR and cooled 4 months was made. The stripping loss was < 0.1% whereas the extraction loss was about 12%. The Th concentration in the TBP was 0.05 mg/ml at equilibrium, and the Pa^{233} decontamination factor was 10^5. Approximately 25 mg of U^{233} were isolated. (auth)

8342 CF-52-11-98(Del.)
Oak Ridge National Lab., Tenn.
PLUTONIUM RECOVERY FROM THE CTF PROCESS RESIDUE. Quarterly Report [for] August 10–November 10, [1952]. J. R. Flanary and E. B. Sheldon. Nov. 14, 1952. Decl. with deletions Feb. 14, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A study was made of the feasibility of using TBP extraction for processing the dissolved fluorine residue of the fluoride volatility processes to recover Pu. (T.R.H.)

8343 CF-53-2-135
Oak Ridge National Lab., Tenn.
MONTHLY PROGRESS REPORT FOR METAL RECOVERY PLANT. R. E. Brooksbank, J. L. Matherne, and W. H. Lewis. Feb. 16, 1953. Decl. Feb. 15, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The operating efficiency of the Metal Recovery Plant for the month of January 1953 is summarized. (W.L.H.)

8344 CF-53-2-164
Oak Ridge National Lab., Tenn.
DESIGN STUDY FOR SLUG HANDLING AND CHARGING FOR THE THOREX PROCESS PILOT PLANT. L. N. Howell, A. M. Tripp, and F. Ring, Jr. Feb. 9, 1953. Decl. Feb. 14, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Studies of several methods of transporting and charging hot slugs for the Thorex Process were made. Hot slugs are stored at the bottom of the pile canal and are to be transported approximately 300 ft and charged into the process dissolvers. Of the methods studied, a combination carrier with an integral feeder arrangement and a stationary metering and selector valve for metering the flow of slugs and directing them to the proper dissolver is favored. Drawings of four proposed methods are given and the characteristics of each are discussed. (D.E.B.)

8345 CF-53-5-20
Oak Ridge National Lab., Tenn.
PRELIMINARY INVESTIGATIONS OF A TBP EXTRACTION PROCESS FOR RECOVERING U^{233} FROM THE THOREX PROCESS CU STREAM. B. G. Ryle. May 4, 1953. Decl. Feb. 14, 1957. 25p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Preliminary studies have been made to aid in the designing of a tributyl phosphate (TBP) extraction process as a possible alternate to the present ion exchange method for purifying and concentrating the U^{233} product stream from the Thorex process. (auth)

8346 CF-53-7-183
Oak Ridge National Lab., Tenn.
TENTATIVE CHEMICAL FLOWSHEET FOR THE DECONTAMINATION AND RECOVERY OF U^{235} FROM IRRADIATED URANIUM–ALUMINUM ALLOYS USING A 1.5% TBP SYSTEM. A. T. Gresky. July 24, 1953. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $.180(ph OTS); $1.80(mf OTS).

8347 CF-54-7-196(Del.)
Oak Ridge National Lab., Tenn.
A SURVEY OF SEPARATION PROCESSES FOR TECHNETIUM. N. A. Krohn. July 29, 1954. Decl. with deletions Feb. 14, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8348 CF-54-8-125
Oak Ridge National Lab., Tenn.
THOREX: PROGRAM FOR ENGINEERING EVALUATION OF THOREX PILOT PLANT EQUIPMENT. A. C. Jealous and E. Lieberman. Aug. 19, 1954. Decl. Feb. 15, 1957. 40p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

A test program is outlined to demonstrate the efficiency and capacity of the pilot-plant-scale equipment for carrying out the various steps of the process and to provide engineering data, including scale-up information, to correlate with earlier Unit Operations experience for future use in designing a Thorex production plant. (F.S.)

8349 CF-54-9-106
Oak Ridge National Lab., Tenn.
ESTIMATION OF RADIATION INTENSITY TO BE EXPECTED IN HR CHEMICAL PROCESSING. H. O. Weeren. Sept. 14, 1954. Decl. Mar. 12, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Calculations of the radiation intensity to be expected in homogeneous reactor chemical processing are briefly summarized. (B.J.H.)

8350 CF-54-9-157
Oak Ridge National Lab., Tenn.
SECOND CYCLE CONSIDERATIONS FOR THE THOREX PROCESS. A. T. Gresky and R. P. Wischow. Sept. 17, 1954. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Various methods for attaining additional decontamination from ruthenium of the Thorex thorium product are discussed as well as the need for such methods. A thorium oxalate precipitation is recommended for obtaining the required additional decontamination in the necessary cases in preference to a solvent extraction system, thorium fluoride precipitation, or thorium nitrate recrystallization. (auth)

8351 CF-55-1-193
Oak Ridge National Lab., Tenn.
A THEORY FOR THE DESIGN OF DISSOLVER FOR SLUG-TYPE REACTOR FUELS. S. H. Jury. Jan. 25, 1955. Decl. Feb. 15, 1957. 18p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The theory and design equations for dissolvers for spent slugs are developed. A comparison with experimental results is still in progress, however, preliminary findings indicate good agreement. (D.E.B.)

8352 CF-55-2-92(Del.)
Oak Ridge National Lab., Tenn.
25 PROCESS ASSISTANCE FOR PERIOD OF JANUARY 21, 1955 TO FEBRUARY 4, 1955. F[loyd] L. Culler. Feb. 15, 1955. Decl. with deletions Feb. 14, 1957. 8p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Laboratory batch dissolutions of cast CF U alloy slugs, chemistry studies of 1A column, production of $ThCl_4$ by chlorination of Th oxalate, reduction of $ThCl_4$ with sodium amalgam, dissolution of $ThCl_4$ in PDA, and hot pressing of Th quasi-amalgam are briefly discussed. (J.E.D.)

8353 CF-55-2-172(Del.)
Oak Ridge National Lab., Tenn.
25 PROCESS ASSISTANCE FOR PERIOD OF FEBRUARY 4, 1955 TO FEBRUARY 19, 1955. F[loyd] L. Culler. Feb. 29, 1955. Decl. with deletions Feb. 14, 1957. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

To prevent contamination from handling and to facilitate dissolution of LITR fuel plates, a dissolver is proposed which accommodates full length assemblies. Some 25 Process flowsheet changes and continuous normal U dissolution are discussed. Some Metallex Process studies are reported and decontamination of the Thorex Pilot Plant is discussed. (T.R.H.)

8354 CF-55-7-107
Oak Ridge National Lab., Tenn.
METALLEX DRY-SALT PROCESS DEVELOPMENT REQUIREMENTS. J. C. Bresee and J. C. Suddath. July 26, 1955. Decl. Feb. 15, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

All dry salt process flow sheets under active consideration or consideration as alternates for the Metallex Process were evaluated to aid in coordinating process development. (F.S.)

8355 CF-55-8-180(Del.)
Oak Ridge National Lab., Tenn.
THOREX PILOT PLANT RUN HD-8 SUMMARY. W. T. McDuffee, C. V. Ellison, and W. R. Whitson. Aug. 5, 1955. Decl. with deletions Feb. 14, 1957. 58p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60(mf OTS).

The performance of the Thorex Process operating on flowsheet No. 1.2-7 is reported. Thorium decontamination by a factor of 1.6×10^4 is achieved with a 98.7% recovery. The U^{233} product was decontaminated by a factor of 3.3×10^4 with a recovery of 97.8% before sorption. Process performance and efficiency are discussed, and the conclusions reached are listed. (T.R.H.)

8356 CF-56-1-3
Oak Ridge National Lab., Tenn.
RAW MATERIALS ECONOMIC STUDIES: A PRELIMINARY ESTIMATE OF THE COST OF URANIUM RECOVERY USING AMEX PROCESS WITH THE CLARIFIED LIQUOR CONTAINING 0.1% SOLIDS. B. B. Klima, R. R. Wiethaup, and R. H. Guymon. Jan. 3, 1956. 13p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The cost of uranium recovery from clarified leach solution containing 0.1% solids using the Amex Process is reported. (auth)

8357 CF-56-2-157
Oak Ridge National Lab., Tenn.
THOREX PROCESS SUMMARY, FEBRUARY 1956. A. T. Gresky. Apr. 25, 1956. Decl. Feb. 14, 1957. 201p. Contract [W-7405-eng-26]. $31.80(ph OTS); $9.30(mf OTS).

The Thorex process has the purpose of separating and decontaminating Th, U^{233}, and Pa^{233} from neutron-irradiated Th. Designed principally for the processing of Al-clad Th slugs, the process may be modified for use with other reactor materials, such as ThO_2 or $ThOCO_3$. It is a solvent-extraction process which employs TBP as the extractant, HNO_3 as the Th-dissolution agent, and either $Al(NO_3)_3$ or HNO_3 as the salting agent. Final isolation of the U^{233} product is accomplished by ion exchange. (auth)

8358 CF-56-3-154
Oak Ridge National Lab., Tenn.
THOREX PILOT PLANT: DECONTAMINATION OF THE FEED ADJUSTMENT SYSTEM AND RADIATION EXPOSURES RECEIVED BY PERSONNEL ENGAGED IN MODIFYING A VAPOR LINE IN THE SYSTEM. J. H. Walker and G. S. Sadowski. Mar. 27, 1956. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The feed adjustment tank, the feed tank, acid fractionator, extraction column, and aqueous waste surge tank were decontaminated with several treatments each of 30% nitric acid and 20% sodium hydroxide-2% sodium tartrate. The phase separator was shielded with lead. Eleven maintenance personnel worked in the all a total of 74 man-hr., receiving a total of 1715 mr, an average dosage rate of 23 mr/hr. (M.H.R.)

8359 CF-56-5-100(Del.)
Oak Ridge National Lab., Tenn.
FEED MATERIALS PROCESSING STATUS REPORT FOR MAY 15, 1956. R. E. Blanco and W. K. Eister. May 15, 1956. Decl. with deletions Feb. 14, 1957. 26p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Developmental studies were continued on Metallex, Fluorox, and Excer Processes. Cost estimates of the Metallex Process and equipment descriptions are included. (C.W.H.)

8360 CL-P-374
Clinton Labs., Oak Ridge, Tenn.
SEMIMONTHLY REPORT AND MANPOWER DISTRIBU-

TION FOR PERIOD DECEMBER 15-31, 1944. S. G. English. Jan. 9, 1945. Decl. Feb. 16, 1957. 17p. Contract [W-7405-eng-39]. $3.30(ph OTS); $2.40(mf OTS).

Studies pertaining to the development of the Bismuth Phosphate Process for Pu recovery are summarized. (D.E.B.)

8361 CN-1884

Clinton Labs., Oak Ridge, Tenn.

CHEMISTRY DIVISION SEMIMONTHLY REPORT FOR THE PERIOD ENDING AUGUST 15, 1944. Warren C. Johnson. Decl. Feb. 16, 1957. 17p. Contract [W-7405-eng-39]. $3.30(ph OTS); $2.40(mf OTS).

8362 CN-2279

California. Univ. Berkeley. Radiation Lab.

THE THORIUM OXALATE EXTRACTION-DECONTAMINATION PROCESS. Oct. 5, 1944. Decl. Feb. 18, 1957. 41p. Contract [W-7401-eng-37]. $7.80(ph OTS); $3.30(mf OTS).

Studies were performed to enable the selection of optimum conditions for the carrying of Pu(IV) by Th when it is precipitated by acidifying an ammonium oxalate solution. The exchange of Th with $Th_2(C_2O_4)_5(NH_4)_3 \cdot 7H_2O$, and the pickup of Pu by the same precipitate were investigated using both HNO_3 and H_2SO_4 to precipitate the Th. It was found in all cases that Th exchanged more rapidly than Pu was picked up, and that both rates were greater at 75°C than at room temperature. Both the rate of exchange of Th and of pickup of Pu(IV) appeared to be slower in H_2SO_4 than in HNO_3. A study has been made of the solid which precipitates when $La(NO_3)_4$ is added to an ammonium oxalate solution of Th. The solid contains La, Th, ammonium and oxalate. The mole ratio of La to Th in 0.3 $\underline{M}$ $(NH_4)_2C_2O_4$ was found to be 3 to 2. The approach to equilibrium was even slower if the Th and La were first precipitated as mixed oxalates and then allowed to digest in ammonium oxalate solution. The formula of the mixed compound at equilibrium is believed to be $La_3Th_2(NH_4)_8 \times (C_2O_4)_{12.5} \cdot 16H_2O$. Pu losses when La was precipitated out of ammonium oxalate solution containing Th were determined. It was found that Pu(IV), like Th, was picked up only very slowly by the La precipitate. However, losses of 2 to 4% of the Pu were obtained in such a precipitation. Variation of the ammonium oxalate and Th concentrations gave Pu losses of the same magnitude or greater. A series of mixed oxalates containing La, Th and ammonium were analyzed for C and H_2 by the standard Pregel micro C–H analysis and for ammonium ion by the micro Kjeldahl method. A diagram and a description of the micro Kjeldahl apparatus is given. Analyses of known amounts of ammonium sulfate indicate that the determination of ammonium ion is reliable to about 2%. (auth)

8363 CN-2564

Clinton Labs., Oak Ridge, Tenn.

STUDIES ON THE CARRYING OF COLUMBIUM(V) [NIOBIUM] BY BISMUTH PHOSPHATE PRECIPITATES. Final Report [for] June 1, 1944 to January 1, 1945 [on] Problem Assignment 203-X72C. Part VI. A. R. Brosi. Feb. 1, 1945. Decl. Feb. 19, 1957. 51p. Contract W-7405-eng-39. $9.30(ph OTS); $3.60(mf OTS).

The carrying of radiocolloidal and soluble Nb by $BiPO_4$ precipitates was studied. At concentrations where Nb is soluble ($< 10^{-4}$ M) it is carried within the $BiPO_4$ crystals with a Nernst distribution constant of 10. Preliminary experiments on the rate of incorporation indicate that the rate can be related to rate of exchange of RaE and active PO_4^{3-} with solid $BiPO_4$. (auth)

8364 DP-25

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.

TNX EVAPORATOR INCIDENT JANUARY 12, 1953. Interim Technical Report. T. J. Colven, Jr., G. M. Nichols, and T. H. Siddall. May 15, 1953. Decl. Feb. 16, 1957. 35p. Contract AT(07-2)-1. $6.30(ph OTS); $3.00(mf OTS).

The circumstances are described pertaining to an explosion in a TNX evaporator during the concentration of a uranyl nitrate-nitric acid solution. TBP and Amsco diluent were probably present in the evaporator charge. A laboratory program involving small-scale atmospheric and closed-vessel distillations of various mixtures of UNH, HNO_3, Amsco, TBP, and water indicate that an exothermic reaction between TBP and UNH, or between TBP and HNO_3, or both, was responsible for the explosion. The danger areas of temperature, TBP concentration, pressure, and heating rate are defined and recommendations are made for safe evaporator operation. (auth)

8365 HDC-1840

[Hanford Works, Richland, Wash.]

WASTE METAL RECOVERY PROCESS. Technical Data on Project C-362. H. G. Johnson and T. I. Peterson. July 31, 1950. Decl. Feb. 20, 1957. 31p. Contract [W-31-109-eng-52]. $6.30(ph OTS); $3.00(mf OTS).

Technical data pertaining to the physical properties of chemicals used in the TBP Process are compiled. These chemicals include the HNO_3-H_2O, $Na-OH-H_2O$, $Na-NO_3-H_2O$, $NH_2SO_3H-H_2O$, $UO_2(NO_3)_2-H_2O$, $UO_2(NO_3)_2-H_2O-HNO_3$, and $UO_2(NO_3)_2-H_2O-TBD-DeoBase$ systems. (B.J.H.)

8366 HW-10620

Hanford Works, Richland, Wash.

REDOX EQUIPMENT TESTING REPORT NO. 4. PROGRESS REPORT [FOR] PERIOD JUNE 15 TO JULY 31, 1948. V. R. Cooper and A. W. Allen. Aug. 9, 1948. Decl. Feb. 19, 1957. 27p. $6.30(ph OTS); $3.00(mf OTS).

Testing of the G.E. and C.L. (General Engineering and Consulting Laboratory) Turbine Pump No. 1 was concluded. After 1008 hr of life test operation, the pump was dismantled and critical parts were weighed and measured. Appreciable wear of the rotating elements and mating parts was observed. A block and two pistons simulating construction of a pump (manufactured by Nathan Aircraft Devices, Inc.) was received. The block was submerged in IAF (feed solution) and one of the pistons was diven by a spring-loaded cam actuated by a small electric motor at the rate of 800 cycles/min through a stroke of approximately $\frac{1}{4}$ in. for 147 hr. The other piston was placed in the same solution to undergo a static test for the same time period. Weights and measurements of the pistons are tabulated as taken before and after the test. Performance characteristics of the Design Engineering Diaphragm Pump have been obtained using water as a test fluid and $\frac{1}{16}$-in. polyethylene and $\frac{3}{64}$-in. MFP-10 (monochlor-tri-fluor-polyethylene) diaphragms. Performance data have been obtained. Diaphragm failure, faulty valve operation, and the presence of air in the bellows actuating systems were the principle difficulties encountered during the test. (D.E.B.)

8367 HW-13481

Hanford Works, Richland, Wash.

REDOX: ANALYTICAL RESEARCH GROUP REPORT [FOR] MARCH AND APRIL 1949. H. R. Schmidt. June 21, 1949. Decl. Feb. 27, 1957. 36p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf OTS).

8368 HW-16076

Hanford Works, Richland, Wash.

CHEMICAL RESEARCH SECTION PROGRESS REPORT FOR JANUARY 1950. F. W. Albaugh. Feb. 15, 1950. Decl. Feb. 19, 1957. 36p. $6.30(ph OTS); $3.00(mf OTS).

8369 HW-17937

Hanford Works, Richland, Wash.

RECOVERY OF NITRIC ACID IN THE TRIBUTYL

PHOSPHATE PROCESS FOR URANIUM RECOVERY. PART 2. A. CORROSION STUDIES. B. PATH OF CHLORIDE ION. C. PATH OF TRIBUTYL PHOSPHATE. R. M. Wagner and C. Groot. May 29, 1950. Decl. Feb. 19, 1957. 12p. Contract W-31-109-eng-52. $0.25(OTS).

The corrosion rates for type 316 stainless steel are given for various parts of the distillation system. The path of TBP in distillation of column waste (RAW) and the path of chloride during RAW concentration and HNO_3 distillation are outlined. (T.R.H.)

8370 HW-19023

Hanford Works, Richland, Wash.

SCOUTING STUDIES IN A 5.05-INCH DIAMETER REDOX PULSE COLUMN. W. S. Figg and J. G. Bradley. Jan. 15, 1951. Decl. Feb. 19, 1957. 21p. Contract W-31-109-Eng-52. $6.30(ph OTS); $3.00(mf OTS).

Since studies in a 1-in. glass column had indicated that savings in column length of the stainless steel, Raschig-ring-packed, liquid-liquid redox production plant contactors might be realized if pulse columns were employed, a series of pilot-plant pulse-column scouting studies have been carried out. The objectives of these pulse-column scouting studies were to scout the extraction performance of a 5-in.-diam pulse column operating as a IA simple extraction section, a IB simple scrub section, and a IC column; to make only a limited number of runs (24) simulating each column to determine the flooding and extraction performance for comparison with 1-in. pulse column and plant-scale packed column performance; to evaluate performance using only one column geometry arbitrarily selected. Complete data for these runs are tabulated, and new information obtained is summarized. (C.H.)

8371 HW-19065

Hanford Works, Richland, Wash.

COMPARISON OF DILUENTS FOR TRIBUTYL PHOSPHATE. L. L. Burger. Oct. 6, 1950. Decl. Feb. 20, 1957. 22p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40 (mf OTS).

A study of the composition and physical properties of a number of hydrocarbon diluents with some generalizations regarding relations between these properties is summarized. Tabular data on the following are included: physical properties of pure diluents; physical properties of typical pure hydrocarbons; flash points of TBP-diluent mixtures; properties of diluent-TBP mixtures saturated with H_2O at 25°C; and comparison of Stoke's factors for metal recovery systems and for purex systems. Curves for ASTM distillation of hydrocarbon diluents of various kinds, viscosity-flash point relation for hydrocarbon mixtures, and viscosity of organic phase containing $UO_2(NO_3)_2$ at 25°C are included. 18 references. (C.H.)

8372 HW-19112

Hanford Works, Richland, Wash.

RESULTS OF CRITICAL MASS STUDIES FOR HANFORD AT K-25. Paul F. Gast. Oct. 11, 1950. Decl. Feb. 19, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

Results of critical mass studies for the Bismuth Phosphate Separation Process at Hanford show that iron, in the thickness existing in process equipment, is a less effective tamper than water, the moderating effect of Bi and P ion is inferior to that of the H_2O which they displace, and their presence in the core or the reflector will not decrease the critical mass, and solutions of uranyl nitrate in concentrations occurring in the separations process are no more effective as tampers than H_2O. (B.J.H.)

8373 HW-20287

Hanford Works, Richland, Wash.

DECONTAMINATION OF REDOX FEED SOLUTIONS BY SUPER FILTROL SCAVENGING. W. E. Roake and H. G. Hicks. Sept. 1, 1950. Decl. Mar. 4, 1957. 20p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The scavenger dissolver solution ratio at which the practical maximum absorption of Nb and Zr is reached was determined, together with the effect of time of prior oxidation of the dissolver solution, dependence on acidity, and effect of consecutive and countercurrent batch scavenging on adsorption of Zr and Nb. (G.Y.)

8374 HW-27821

[Hanford Atomic Products Operation, Richland, Wash.]

CORROSION OF STAINLESS STEELS IN PUREX ACID CONCENTRATORS. Kenneth L. Sanborn. Apr. 23, 1953. Decl. Feb. 18, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The corrosion resistance of four types of austenitic stainless steels to Purex process first and second acid concentrator bottoms was determined. Although the compositions of the two solutions were very similar, tests in simulated solutions of both compositions were conducted to strengthen the basis for the conclusions which were to be made. Type 309SCb (sensitized) and Carpenter 20 (annealed) each suffered corrosion rates of 0.0003 in./month. Type 304L (sensitized) suffered a corrosion rate of 0.0009 in./month. Type 316L (sensitized) suffered corrosion rates ranging from 0.0007 to 0.0012 in./month, with evidence of excessive intergranular attack. (auth)

8375 HW-29001

[Hanford Works, Richland, Wash.]

EFFECT OF TEMPERATURE ON URANIUM RECOVERY COLUMN OPERATION. L. L. Burger. Aug. 12, 1953. Decl. Feb. 20, 1957. 7p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Some data on the temperature effect in pulse columns are reviewed. New data obtained with the vibrational stirrer are reported. The effects on both the dispersion and coalescence and the net rate of extraction suggest that improved operation of the RA column and lowered waste losses would be expected at elevated temperatures. (auth)

8376 HW-31300

General Electric Co. Hanford Atomic Products Operation, Richland, Wash. THE VOLATILIZATION OF RUTHENIUM IN NITRIC ACID RECOVERY. Archie S. Wilson. Sept. 3, 1954. Decl. Feb. 25, 1957. 3p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Evidence points toward RuO_4 as being the volatile Ru species. The most important factor in this distillation is the exposure of heating surfaces to still vapor. The prevention of RuO_4 formation by HNO_3 concentration control and minimized still-wall temperatures are suggested as preventative measures. (D.E.B.)

8377 HW-33148

Hanford Atomic Products Operation, Richland, Wash.

pH MONITORING OF METAL RECOVERY NEUTRALIZED WASTE. C. E. Michelson. Sept. 24, 1954. Decl. Feb. 25, 1957. 23p. Contract W-31-109-Eng-52. $0.35(OTS).

In order to ensure adequate control of pH in the continuous neutralization of Metal Recovery Process wastes, a continuous recording pH meter has been successfully applied. The mechanical difficulties of a pH sensing system working in a relatively inaccessible location with a continuously operated jet sampler have been overcome using a replaceable unit with provisions for standardizing the pH meter and automatically acid-flushing the cell and sampler. A unique feature of the unit is that the acid flush which is introduced at regular intervals provides, by means of the pH meter response, a positive indication on the recorder chart that both the sampler and the pH meter are function-

ing properly. The close pH control made possible by this installation has been shown to be adequate for the successful scavenging of radiocesium from process wastes and has also reduced the frequency of over-neutralization of the waste. (auth)

8378 IDO-14216
American Cyanamid Co. [Atomic Energy Div.], Idaho Falls, Idaho.
PRODUCT PURITY AND CORROSION IN CPP INTERCYCLE EVAPORATORS. W. J. Guay. Feb. 11, 1953. Decl. Feb. 27, 1957. 7p. Contract AT-(10-1)-177. $1.80 (ph OTS); $1.80(mf OTS).

The contribution of impurities from corrosion is small. Impurities of Al and Si are greater than expected. Part of the Si is from the 10 ppm Si in demineralized water. The use of distilled water would cut this down. Fe and Na content are acceptably low. The corrosion rate in the product evaporator is 0.75 mil/yr and in the strip columns it is 1.5 mil/yr. (F.S.)

8379 K-817
Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.
RECOVERY OF PLUTONIUM AND FISSION PRODUCTS FROM REACTOR POT RESIDUES OF THE CHLORINE TRIFLUORIDE PROCESS. S. T. Benton and R. A. Gustison. Sept. 12, 1951. Decl. Feb. 23, 1957. 16p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Preliminary investigations have shown that the Pu and fission-product fluoride residue can be recovered quantitatively from the reactor pot of the CTF Process without prohibitive corrosion to the material of construction. Aluminum nitrate was found to be superior to Versene as a solvent in removing the Pu from a prototype reactor, but Versene more thoroughly removed the fission products, especially Ru. The maximum solubility of PuF_4 was 23.2 g/l in 5% $Al(NO_3)_3$ at pH 0.5 and 4.1 g/l in 1% Versene at pH 9. Monel was satisfactory as a material of construction in either $Al(NO_3)_3$ of Versene with a rate of corrosion of 0.022 mils/day and 0.001 mils/day respectively. Data indicates that Ni would not be very satisfactory for use at the optimum solubility conditions of 5% $Al(NO_3)_3$, since the rate of corrosion was 0.366 mils/day. (auth)

8380 KAPL-M-Redox-2
Knolls Atomic Power Lab., Schenectady, N. Y.
REDOX GROUP MONTHLY REPORT FOR OCTOBER 1948. Oct. 26, 1948. Decl. Feb. 28, 1957. 30p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

A monthly report was made on the progress in the Redox Group on heavy-element chemistry, fission-product chemistry, and analytical methods. (F.S.)

8381 KLX-1607
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING PROGRESS REPORT [FOR] PERIOD NOVEMBER 1–30, 1952. Job 57. Dec. 10, 1952. Decl. Mar. 1, 1957. 30p. Contract AT (11-1)-217. $4.80(ph OTS); $2.70(mf OTS).

8382 KLX-1720
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR FUEL REPROCESSING. Quarterly Progress Report for January 1–March 31, 1954. Job 1087. Apr. 19, 1954. Decl. Mar. 1, 1957. 21p. Contract W-7405-Eng-26, Subcontract 535. $4.80(ph OTS); $2.70(mf OTS).

The development of a simple process for separating rare-earth fission products from short-cooled UO_2SO_4 fuel solutions is reported. The solubility of individual and mixed rare-earth sulfates in homogeneous reactor fuel at 250 to 325°C, the evaluation of a laboratory and pilot-plant process for rare earth adsorption by CaF_2, methods for removing dissolved Ca and fluoride from adsorber effluent, and an economic study of the CaF_2 adsorption process are reported. Possible methods of separating insoluble fission products from homogeneous reactor fuel at 270 to 300°C were studied as alternatives to adsorption by CaF_2. (J.E.D.)

8383 KLX-1722
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR FUEL REPROCESSING. Quarterly Progress Report [for] April 1–June 30, 1954. Job 1087. July 30, 1954. 17p. For Oak Ridge National Lab. Contract W-7405-eng-26, Subcontract 535. $3.30 (ph OTS); $2.40(mf OTS).

Solubility studies were made in static bomb tests of neodymium, cerium, and strontium sulfates in 0.02M UO_2SO_4 (with 25% excess H_2SO_4), at 250 to 325°C. Also, similar solubility tests were made to determine the carrying effect of the coprecipitation of rare earths. These tests were made with cerium, neodymium, and europium plus radioactive tracers in the presence of gross quantities of mixed rare earths. In these solubility studies all rare earths exhibited pronounced decrease in solubility with temperature. $Ce_2(SO_4)_3$ was the least soluble, and its solubility was lowered the most by the presence of other rare earths. A study of neodymium tracer reliability indicated that not over 11% of its radioactivity was protactinium. Bomb tests are being made to study the behavior of iodine in 0.02M UO_2SO_4, at reactor temperatures. Emphasis is being given the interaction of iodine with stainless steel. The second alternative process for CaF_2 absorber effluent post-treatment has been revised. It was learned that simultaneous precipitation of ThF_4 and $CaSO_4$ as a single operation at 275° would give low fluoride removal, due to the tendency of thorium to precipitate as the sulfate at that temperature. Therefore, the process flow sheet and cost analysis were amended to include ThF_4 removal at 100°C before the high-temperature $CaSO_4$ precipitation in either (1) two zones of a cartridge or (2) in two separate cartridges, at a very small additional cost. Precipitation of ThF_4 by adding anhydrous $Th(SO_4)_2$ to a 1-liter batch of adsorber effluent and filtering the resulting slurry in a column-type sand-bed was demonstrated. (For preceding period see KLX-1720.) (J.S.R.)

8384 LWS-22654
California Research and Development Co., Livermore, Calif.
APPLICATION OF AQUEOUS PROCESSES TO FBR BLANKET PROCESSING. M. L. Feldman. June 2, 1953. Decl. Feb. 27, 1957. 21p. (D-T-16). $4.80(ph OTS); $2.70(mf OTS).

Aqueous processing can be simplified considerably for processing FBR blanket material. It is also concluded that in order to achieve the optimum in an aqueous process simplification, a UO_2 blanket and a Pu salt or dioxide core should be used. It is recommended that a program be launched to develop an aqueous process utilizing adsorption and precipitation techniques, either alone of in combination. It is also recommended that a process be investigated using a Purex IA column followed by the precipitation of Pu and U from the organic extractant. (auth)

8385 M-4549
Linde Air Products Co. Tonawanda Lab., Tonawanda, N. Y.
PROCESSING OF AFRICAN ORES. E. O. Brimm. May 7, 1943. Decl. Mar. 6, 1957. 13p. $3.30(ph OTS); $2.40 (mf OTS).

The processing of high-grade and low-grade U ores from

pitchblende deposits for recovery of U and Ra was investigated. High U recoveries were obtained by an H_2SO_4 leaching of the ore followed by an alkaline digestion, the Ra remaining in the residue. A technique was developed for removing Mo and Pb from the leach liquors. (C.W.H.)

8386 NAA-SR-245
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JANUARY–MARCH 1953. E. Motta and M. Yardley, eds. May 8, 1953. Decl. Mar. 2, 1957. 56p. Contract AT-11-1-Gen-8. $9.30(ph OTS); $3.60(mf OTS).

High-temperature compact separations methods for nuclear fuels. Experimental data for the rate of evaporation of tracer Pu from molten U were found to agree with theoretical values at 1600 and 1680°C. Theoretical calculations indicated feasibility of large-scale Pu evaporation. The emissivity of molten U in a vacuum was determined to have a value of 0.82. Evaluation of furnace assemblies for the scale-up 3-in.-diam. vacuum system to process kilogram weights of U indicated that Carbocell would be unsatisfactory as a component above 1000°C and that zirconia was a suitable U container up to 1400°C. Two experiments in which molten U was scrubbed with fused UF_4 yielded decontamination data for Pu, Zr, Nb, Ru, and gross β-γ. The data indicated about 2 hr as the practical limit for a single scrub, and multiple-cycle scrubbing appeared promising as a means of increasing decontamination. Adaptation of the fluoride volatility process to various fuel mixtures. The removal of 99.9% of the U from sintered UO_2-MgO mixtures was accomplished by treatment with BrF_3. A new analytical method for the quantitative separation and recovery of U from BrF_3-Br_2 mixtures was developed. More than 99% of the U was separated from "Funak" (NaF-KF-UF_4) fused-salt fuel by treatment with a BrF_3 procedure. "Funazr" (NaF-ZrF_4-UF_4) fuel proved resistant to direct BrF_3 treatment for U separation as UF_6; however, pretreatment by fusion with KF resulted in good U recovery from "Funazr" by reaction with BrF_3. Radiation chemistry of the compact high-temperature processes and the fluoride volatility process. Heavy irradiation of BrF_3 with 1-Mev electrons indicated a radiation effect that approached a steady state. The presence of Br_2 and BrF_5 as radiation decomposition products of BrF_3 was demonstrated. Irradiations of a BrF_3-U system with 1-Mev electrons yielded data indicating an increase in the reaction due to the radiation. Irradiation of US_2 with 1-Mev electrons resulted in formation of US and elemental S. (auth)

8387 NAA-SR-273
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY PROGRESS REPORT FOR PERIOD APRIL, MAY, JUNE 1953. E. E. Motta, D. W. Bareis, and D. D. Cubicciotti. Sept. 24, 1953. Decl. Mar. 2, 1957. 34p. Contract AT-11-1-GEN-8. $6.30(ph OTS); $3.00(mf OTS).

Encouraging results have been obtained in the removal of Pu and fission products from molten U metal by simple distillation and by fused fluoride extraction. A combination of the above methods utilizing the higher volatility of the fused salts has also been investigated. Separation of Np from molten U by fused halide extraction and volatilization has been tried. Certain fused salts were effective in dissolving reactor fuel alloys such as Al–U and Zr–U, which allowed the use of the fluoride volatility process for the recovery of the uranium. Successful operation of the pilot scale BrF_3 distillation equipment has been achieved. Statitron irradiation of BrF_3 has caused decomposition at high radiation intensities. (auth)

8388 NAA-SR-844
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JULY–SEPTEMBER, 1953. E. E. Motta, D. W. Bareis, and D. Cubicciotti, eds. Apr. 1, 1954. Decl. Mar. 4, 1957. 28p. Contract AT-11-1-GEN-8. $0.35(OTS).

Continued progress has been made with the high temperature decontamination processes for irradiated U fuel. The fused salt treatment of molten U has been extended to UCl_3. Plutonium and rare earths were extracted into the UCl_3 phase. Direct Pu distillation from molten irradiated U has been scaled up to the hundred gram scale. Solid scavenging experiments using U oxide, U carbide, and U nitride in contact with molten U have indicated fission product removal. A scaled-up investigation of the separation and recovery of U from an SIR type ceramic fuel using the volatile fluoride process has indicated the feasibility of this separation method. The effect of irradiation on the decomposition of BrF_3 has been further studied in experiments using the NAA statitron. (auth)

8389 NAA-SR-1032
North American Aviation, Inc., Downey, Calif.
THE DISTRIBUTION OF TRACER PLUTONIUM AND FISSION PRODUCTS BETWEEN MOLTEN URANIUM AND SOLID URANIUM OXIDE, CARBIDE, AND NITRIDE. F. J. Keneshea, Jr., A. M. Saul, and C. T. Young. Sept. 15, 1954. Decl. Mar. 12, 1957. 36p. Contract AT-11-1-GEN-8. $0.35(OTS).

A study has been made of the distribution of tracer fission products and plutonium between small samples of molten uranium and solid uranium oxide, carbide, and nitride. The distribution showed the same behavior in general for all three materials. The rare earth elements, Cs, Ba, and Sr, were extracted primarily into the solid scrub phase. Zirconium and Nb partially concentrated in the scrub phase. Plutonium, Mo, and Ru tended to remain completely in the metal phase. The distribution of activities agreed with trends predicted from the thermodynamic data. Uranium oxide appeared to be the most desirable scrub material for removing large amounts of fission products from the uranium while leaving behind the Pu. In addition the uranium metal was not severely contaminated by dissolved oxide. (auth)

8390 NAA-SR-1628
Atomics International Div., North American Aviation, Inc Canoga Park, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] OCTOBER TO DECEMBER 1955. E. Motta, A. Buyers, S. Yosim, and J. Foltz, eds. July 15, 1956. Decl. Mar. 1, 1957. 45p. Contract AT-11-1-GEN-8. $0.45(OTS).

Emphasis has now been placed on the processing of up to 25-kg batches of irradiated U fuel. A 50-kg induction-heated vacuum-melting furnace has been tested by melting and casting normal U using ordinary contact methods. Laboratory-scale experiments on the decontamination of tracer-irradiated Th–U alloy have shown that the de Boer iodide process is effective in removing many of the fission products, including Ru. Volatilization of fission elements from solid Th–U alloy demonstrates that at least Cs, Sr, and Te can be removed from the fuel without melting. Salt-scrubbing experiments, in which molten Th–U–Mn alloys are contacted with fused ThF_4 or $CaCl_2$, reveal that $CaCl_2$ is much more effective in removing fission products. Solid and liquid slagging experiments on Th–U alloy continue. The arc-zone melting of Th–U alloy for decontamination has been completed and the results discussed. (For preceding period see NAA-SR-1529.) (auth)

8391 ORNL-371
Oak Ridge National Lab., Tenn.
DIISOPROPYL ETHER AS A SOLVENT FOR U^{233} SEPARATION FROM THORIUM AND FISSION PRODUCTS IN CONTINUOUS COLUMN OPERATIONS. D. E. Ferguson and R. E. Leuze. June 12, 1950. Decl. Mar. 2, 1957. 19p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A continuous solvent extraction process for the recovery of U^{233} from Th and fission products has been demonstrated in packed countercurrent columns, using diisopropyl ether as a solvent. (auth)

8392 ORNL-374
Oak Ridge National Lab., Tenn.
COMPARISON OF U^{233} CONTINUOUS SOLVENT EXTRACTION PROCESSES. D. E. Ferguson and R. E. Leuze. June 27, 1950. Decl. Mar. 2, 1957. 13p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A comparison of four continuous solvent extraction processes for the recovery of U^{233} from thorium and fission products, using methyl isobutyl ketone, diisopropyl ether and dibutyl cellosolve as solvents is presented. It was concluded that both the diisopropyl ether and methyl isobutyl ketone processes yielded adequate U^{233} separation and recovery. However, additional studies are necessary on stripping dibutyl cellosolve before either of the two proposed processes utilizing this solvent can be used. (auth)

8393 ORNL-890
Oak Ridge National Lab., Tenn.
EVALUATION OF TRIBUTYL PHOSPHATE IN JET MIXING COLUMNS AS A SOLVENT FOR URANIUM EXTRACTION FROM SLURRY-TYPE ORE FEEDS. T. C. Runion and J. H. Yeager. Dec. 7, 1950. Decl. Mar. 2, 1957. 15p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The physical operability of 30% TBP in a hydrocarbon diluent for the extraction of U from ore feed slurries has been satisfactorily demonstrated in jet mixing columns at Mallinckrodt pilot plants. (auth)

8394 ORNL-1364
Oak Ridge National Lab., Tenn.
AN ION-EXCHANGE PROCESS FOR U^{233} ISOLATION AND PURIFICATION. D. C. Overholt. Oct. 4, 1952. Decl. Mar. 2, 1957. 22p. Contract W-7405-eng-26. $0.30(OTS).

An ion-exchange process for coupling the Interim-23 solvent-extraction separation process for U^{233} to the final metallurgical reduction was developed and satisfactorily demonstrated on a full-scale application. The U product was satisfactorily carried through a UO_4 precipitation, which is the initial step in the metallurgical reduction process. A mass analysis of the purified product was made. (auth)

8395 ORNL-1367(Rev.)
Oak Ridge National Lab., Tenn.
LABORATORY DEVELOPMENT OF THE THOREX PROCESS. Progress Report. A. T. Gresky, M. R. Bennett, S. S. Brandt, W. T. McDuffee, and J. E. Savolainen. Dec. 17, 1952. Decl. Mar. 4, 1957. 157p. Contract W-7405-eng-26. $0.85(OTS).

The report presents data and conclusions obtained during preliminary laboratory development of the Thorex Process, which was required to separate and decontaminate Th, U^{233}, and Pa from irradiated Th metal. Tentative chemical flowsheets are described. (auth)

8396 ORNL-1518
Oak Ridge National Lab., Tenn.
LABORATORY DEVELOPMENT OF THE THOREX PROCESS: PROGRESS REPORT, OCTOBER 1, 1952 TO JANUARY 31, 1953. A. T. Gresky, M. R. Bennett, S. S. Brandt, W. T. McDuffee, B. G. Ryle, J. E. Savolainen, and R. P. Wischow. Aug. 4, 1953. Decl. Mar. 12, 1957. 93p. Contract W-7405-eng-26. $13.80(ph OTS); $4.80(mf OTS).

Primary engineering features of the Thorex Process and recent revisions in the chemical flowsheet are indicated. Decontamination of the Th–U stream from Pa in the A-column was shown to be markedly dependent on the presence of phosphate ion. Inclusion of 0.002M H_3PO_4 in the AS stream was found optimum both for control of Pa separation and to permit insignificant Th losses in the AP stream. The inclusion of a neutralization column to remove HNO_3 from the BU stream in order to improve conditions of the CU stream and to permit a more nearly optimum operation of the U^{233} ion-exchange isolation step is discussed. The optimum system for Pa recovery from the AP stream is undecided, and alternate procedures are being investigated and evaluated for preliminary plant design. Adaptation to efficient plant operation of the silica-gel adsorption system is considered doubtful. In addition, certain unknown variables adversely affecting Pa adsorption were encountered, and radiation instability of silica-gel structures under prospective high exposures caused uncertainties regarding its use as a Pa sponge. (auth)

8397 ORNL-1784
Oak Ridge National Lab., Tenn.
THOREX SOLVENT RECOVERY: DEVELOPMENT OF CALCIUM HYDROXIDE SLURRY PROCESS. P. A. Haas and A. C. Jealous. Feb. 16, 1955. Decl. Mar. 2, 1957. 30p. Contract W-7405-eng-26. $0.35(OTS).

Calcium hydroxide slurries, approximately 360 g of calcium hydroxide per liter of water, effectively removed impurities from Thorex process solvent (42% TBP in Amsco 123-15) which had been used in nonradioactive pulse column studies, so that it could be reused in extraction columns. Contact between solvent and slurry was made in mixer settlers. The volume of waste from this recovery process was 1% or less of the volume of solvent treated. An equipment diagram and a flowsheet are given. (auth)

8398 ORNL-1855
Oak Ridge National Lab., Tenn.
THE ISOLATION AND PURIFICATION OF AMERICIUM. D. O. Campbell. Apr. 17, 1956. Decl. Mar. 2, 1957. 25p. Contract W-7405-eng-26. $0.35(OTS).

Gram amounts of Am were separated quantitatively from kilogram quantities of La to yield an Am product approaching 90% purity. The remaining impurity was chiefly Y. Elution of Am from a 25% loaded Dowex 50 resin column with 0.15M citric acid–0.10M diammonium citrate–0.3M ammonium nitrate, pH 3.3, gave a product containing 99% of the Am with a La/Am ratio of 1/100 or less in one fourth of a column volume, in this case about a 100-fold volume reduction. Approximately 9 g of Am was purified by this method. Elution with 12.8M hydrochloric acid from a 20 to 30% loaded column gave 90% of the Am in two column volumes of product with a La/Am ratio of about 1/4. About 1 g of Am was purified by this method. (auth)

Refer also to abstracts 8440 and 8619.

SYNTHESES

8399 A-2123
Columbia Univ., New York. Div. of War Research.
PILOT PLANT CONVERSION OF $(CH_3)_2O{\cdot}BF_3$ TO CRYSTALLINE B. C. A. Hutchison. Nov. 10, 1944. Decl. Feb. 12, 1957. 15p. Contract W-7405-eng-50. (100B-R-240). $4.80(ph OTS); $2.70(mf OTS).

Yield studies were made of the various steps in the conversion of $(CH_3)_2O \cdot BF_3$ to B. The yield of B from the entire process if 56.4% of the B in $(CH_3)_2O \cdot BF_3$. (D.E.B.)

TRANSURANIC ELEMENTS AND COMPOUNDS

8400 ANL-5037(Del.)
Argonne National Lab., Lemont, Ill.
THE ISOLATION AND PURIFICATION OF NEPTUNIUM-237 OBTAINED FROM HANFORD WASTES. J. C. Hindman, Philip Wehner, J. C. Sullivan, and Donald Cohen. May 6, 1953. Decl. with deletions Feb. 6, 1957. 18p. Contract W-31-109-eng-38. $3.30(ph OTS); $2.40(mf OTS).

Np^{237} has been recovered with moderate success from Hanford waste concentrates by a batch solvent extraction process using ferrous sulfanate and tributylamine. Possibility of Np–Pu separation in the initial extraction step, most favorable conditions for quantitative Np recovery, and the best procedure for final purification are discussed. Flowsheets and product analyses are included. (L.T.W.)

8401 ANL-5366(Del.)
Argonne National Lab., Lemont, Ill.
ARGONNE PLUTONIUM HEXAFLUORIDE PROGRAM. John G. Malm and Bernard Weinstock. Sept. 1954. Decl. with deletions Feb. 13, 1957. 14p. Contract W-31-109-Eng-38. $3.30(ph OTS); $2.40(mf OTS).

Preliminary results are reported from a study of the preparation and chemical and physical properties of PuF_6. Data are included on radiation decomposition in the vapor phase, thermal stability, reactivity with materials of construction, purification, chemical composition, and the infrared, Raman, and absorption spectra. (C.H.)

8402 ISC-49
Iowa State Coll., Ames.
PURIFICATION OF PLUTONIUM BY A CHROMATOGRAPHIC METHOD. John A. Ayres. June 14, 1949. Decl. Feb. 26, 1957. 124p. $19.80(ph OTS); $6.30(mf OTS).

The status of the adsorption column procedure (using Amberlite IR-1 in the hydrogen form) for separation and decontamination of plutonium as it was developed at Iowa State College is presented. Starting with a single column for the separation of plutonium from uranium, the method was developed to a three-column procedure giving, in addition, specific decontamination factors of 10^4 to 10^6 with yields of over 99%. A method is outlined whereby the decontamination factor can be increased to any desired figure by the use of additional columns, and flow sheets incorporating this method are included. A "hot lab" suitable for large scale experiments with material of moderately high radioactivity was designed and built. Materials having an activity up to five curies were processed in this laboratory. 39 references. (F.S.)

URANIUM AND URANIUM COMPOUNDS

8403 A-2920
Du Pont de Nemours (E. I) and Co. Jackson Lab., Wilmington, Del.
TENTATIVE DYE WORKS PROCESSES FOR THE PRODUCTION OF URANIUM TRIOXIDE AND URANIUM DIOXIDE. C. W. Maynard. Feb. 24, 1943. Decl. Feb. 13, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Uranium(IV) oxide was prepared from $UO_2(NO_3)_2$ by thermal decomposition of the nitrate to UO_3, and subsequent hydrogen reduction to UO_2. Yields (~100%) were obtained. (C.W.H.)

8404 BMI-270
Battelle Memorial Inst., Columbus, Ohio.
DETERMINATION OF RARE EARTHS IN REFINED URANIUM. R. A. Ewing and A. E. Bearse. June 30, 1953. Decl. Feb. 13, 1957. 24p. Contract AT(30-1)-Gen-228. $0.30(OTS).

Rare-earth analyses were performed on several samples of refined uranium oxide and nitrate, using a modified cellulose column procedure. Rare earths were not detectable in samples of 0.03 shotgun or less. Detection limits were from 0.0005 to 0.015 ppm. Negligible quantities of low cross-section rare earths were found in less pure uranium samples, obtained during start-up operations of the pilot plant at the Feed Materials Production Center, primarily those rare earths of greater than average abundance in nature. No correlation was found between rare-earth content and shotgun values. Rare earths do not appear to be contributing significantly to the neutron absorption of refined uranium. A modified cellulose column procedure, which includes a preconcentration solvent-extraction step, is described. Radioactive tracer tests indicating rare-earth recoveries by the method of >98 per cent are reported. (auth)

8405 BMI-895
Battelle Memorial Inst., Columbus, Ohio.
SURFACE ALLOYING OF URANIUM. H. A. Saller, J. R. Keeler, and E. S. Hodge. Jan. 5, 1954. Decl. Feb. 26, 1957. 18p. Contract W-7405-eng-92. $0.30(OTS).

Several methods of producing a surface layer of corrosion-resistant alloy on U were investigated. Cementation of metal powders of zirconium, molybdenum, niobium, nickel, iron, or chromium on U produced spotty coatings which were rough and tested erratically in boiling water. Reduction of zirconium hydride on U strip resulted in a porous metallic coating, and metallic additions to the zirconium hydride failed to reduce substantially the porosity of the coatings. Other methods of surface alloying, such as formation of the hydride "in situ" with subsequent reduction, and miscellaneous techniques resulted in coatings less desirable than those obtained by cementation or by reduction of zirconium hydride. (auth)

8406 BMI-961
Battelle Memorial Inst., Columbus, Ohio.
DEVELOPMENT OF DIMENSIONALLY STABLE URANIUM ALLOYS. F. R. Shober, L. L. Marsh, and G. T. Muehlenkamp. Oct. 26, 1954. Decl. Feb. 26, 1957. 38p. Contract W-7405-eng-92. $0.35(OTS).

This report describes the dimensional stability characteristics of alpha-rolled alpha-annealed, alpha-rolled beta-treated, and alpha-rolled gamma-treated uranium alloys thermal cycled from 100 to 500 C. The elements alloyed with uranium during this investigation included: carbon, chromium, cobalt, germanium, hafnium, lead, manganese, molybdenum, nickel, niobium, silicon, tantalum, tin, titanium, tungsten, vanadium, and zirconium. All alloys except uranium-carbon contained not more than 5.0 a/o alloying element. (auth)

8407 BMI-975
Battelle Memorial Inst., Columbus, Ohio.
THE REACTIVITY OF URANIUM AND BLACOSOLV. H. D. Hannan, P[aul] D. Miller, and R. S. Peoples. Jan 21, 1955. Decl. Feb. 20, 1957. 12p. Contract W-7405-eng-92. $0.25(OTS).

Blacosolv, trichloroethylene, is used in degreasing uranium metal. Tests were made under possible industrial conditions to determine whether a hazardous reaction between uranium and Blacosolv could occur. Temperature-rise measurements showed that there was no reaction of the

metal, in various states of subdivision, with Blacosolv at temperatures up to 850°F. However, a reaction was observed in tests under which the finely divided metal, wet with solvent, was subjected to a severe impact. (auth)

8408 CF-51-12-171
Oak Ridge National Lab., Tenn.
PRODUCTION OF EXTREMELY FINE PARTICLES OF UO_3 BY SPRAY CALCINING OF UNH. M. L. Feldman. Dec. 28, 1951. Decl. Feb. 14, 1957. 13p. [Contract W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8409 CF-53-5-78(Del.)
Oak Ridge National Lab., Tenn.
METAL RECOVERY PROGRAM — CONVERSION OF $UO_2(NO_3)_2$ TO UF_4 BY ION EXCHANGE AND ELECTROLYSIS. E. D. Arnold and E. O. Nurmi. May 4, 1953. Decl. with deletions Feb. 14, 1957. 22p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The economic and operational feasibility of producing UF_4 from UNH solutions by an ion exchange-electrolytic process in pilot plant scale equipment is studied. Scale-up requirements for a larger plant are established. (T.R.H.)

8410 CF-54-3-31
Oak Ridge National Lab., Tenn.
A SURVEY OF U METAL REDUCTION. O. C. Dean. Mar. 4, 1954. Decl. Feb. 15, 1957. 13p. Contract [W-7405-eng-26]. $0.30(OTS).

Recent developments in U metal reduction procedures are outlined. Flowsheets are included for the following processes: Direct oxide, chloride, Green Salt Reduction, and Metallex. (W.L.H.)

8411 CF-54-4-229
Oak Ridge National Lab., Tenn.
FEED MATERIALS SEMI-ANNUAL REPORT [FOR PERIOD ENDING] MARCH 30, 1954. O. C. Dean. Apr. 21, 1954. Decl. Feb. 15, 1957. 22p. Contract W-7405-eng-26. $0.35(OTS).

Progress is noted on the following investigations: reduction of UCl_4 with lithium amalgam; preparation of $ThCl_4$ by chlorination of oxide and oxalate and reduction of $ThCl_4$ with lithium amalgam; electrolytic reduction of thorium compounds and uranium compounds; and production of UF_4 from UO_2F_2. (C.W.H.)

8412 CF-55-4-40
Oak Ridge National Lab., Tenn.
LABORATORY STUDY OF DENITRATION AND REDUCTION IN THE MOVING BED PROCESS. N. A. Krohn. Apr. 6, 1955. Decl. Feb. 15, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The process is shown feasible by the lab studies. Pellets for the bed are prepared by molding or extruding a paste of powdered anhydrous UO_3 + $UO_2(NO_3)_2.6H_2O$ (molten), and are denitrated by a flow of air between 300 and 600°C. Plugging occurs below 150°C. Reduction to UO_2 occurs with H_2 or NH_3 at 300 to 600°C. The only loss in physical strength of the pellets occurs in the reduction, but this is not harmful. (T.R.H.)

8413 CF-55-5-98
Oak Ridge National Lab., Tenn.
TRIP REPORT — FEED MATERIALS PROCESSING CENTER. W. K. Eister and J. E. Moore. May 16, 1955. Decl. Feb. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Criteria of tests for the production of UF_4 from UO_3 in the FMPC moving-bed pilot plant by the Fluorox Process are outlined. (C.W.H.)

8414 FMPC-497
National Lead Co. of Ohio, Cincinnati.
DEGREASING AND PICKLING OF URANIUM CHIPS. B. G. Ryle, R. W. Vest, and J. F. Blum. Dec. 1, 1954. Decl. Feb. 23, 1957. 18p. Contract AT(30-1)-1156. $0.30 (OTS).

Results are presented from a series of laboratory studies undertaken to determine the conditions most favorable for the degreasing and pickling of uranium chips and turnings prior to briquetting. The relative efficiency of detergents and an organic solvent (trichloroethylene) as degreasing agents was investigated. The effects of temperature, nitric acid concentration, and immersion time were studied to determine the conditions necessary to obtain the most complete oxide removal with the minimum loss of uranium metal. Drying conditions most favorable for the prevention of reoxidation were ascertained. (auth)

8415 HW-18497
[Hanford Works, Richland, Wash.]
LABORATORY SURVEY OF DEOXIDANTS FOR URANIUM CHIPS. W. T. Kattner. Aug. 3, 1950. Decl. Feb. 19, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Six different solutions of various concentrations (10% K_2CO_3 – 10% H_2O_2, 5% Na_2O_2, 5% HNO_3 in glacial acetic acid, NH_4CO_3 – 0.5% H_2O_2, NH_4OH – 5% H_2O_2, H_2SO_4 – H_2O_2, 50% HNO_3) were used to deoxidize U chips on a laboratory scale. The performance of these deoxidants is compared, and the most suitable one is selected. (L.T.W.)

8416 HW-26762
Hanford Works, Richland, Wash.
A RAPID METHOD FOR DETERMINING PREFERRED ORIENTATION IN ROLLED URANIUM RODS. M. J. Sanderson, W. V. Cummings, Jr., and R. P. Hurst. Jan. 7, 1953. Decl. Mar. 4, 1957. 12p. Contract [W-31-109-Eng-52]. $0.25(OTS).

The crystallographic orientation existing in rolled U rods may be determined by comparing the diffraction pattern obtained from an unknown specimen with the diffraction pattern obtained from a sample known to be randomly oriented. Although the method may be used to plot a complete pole figure, if such is desired, it has been specifically devised for the study of preferred orientation existing in the rolling and cross-rolling directions. (auth)

8417 HW-33006
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
CHEMICAL FLOWSHEET — PREPARATION OF URANIUM OXIDE (UO_3) FROM URANYL NITRATE HEXAHYDRATE (UNH). R. E. Smith. Sept. 3, 1954. Decl. Feb. 25, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

8418 HW-38686
Hanford Atomic Products Operation, Richland, Wash.
PROPOSED EVALUATION OF AN ALTERNATE UF_4 PROCESS. M. J. Sanderson. Aug. 15, 1955. Decl. Feb. 26, 1957. 3p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

An evaluation was proposed for a method for processing uranyl nitrate solutions for UF_4 by simple precipitation of the reducible double salts $UCaF_6$ and $UNaF_5$ from uranyl nitrate solutions at room temperatures. (F.S.)

8419 K-98
Carbide and Carbon Chemicals Corp., Oak Ridge, Tenn.
THE ROUTINE LABORATORY PREPARATION OF URANIUM HEXAFLUORIDE BY DIRECT FLUORINATION OF URANIUM OXIDE. F. W. Hurd and J. LaGraff. Dec. 12, 1947. Decl. Feb. 7, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

A method is described for the routine preparation of

gram amounts of UF_6 by the direct fluorination of uranic oxide or urano-uranic oxide. This method is used routinely for the preparation of synthetic isotopic standards for the mass spectrometer. Over 100 fluorinations have been made with this method, with an average yield of over 90%. The fluorinations are carried out in an all metal system at 350 to 400°C. The UF_6 is collected in a Ni cold trap cooled with a dry ice-trichloroethylene mixture. When the fluorination is complete, the UF_6 is transferred to a Hoke tube for mass spectrometer use or to a storage cylinder. The fluorination of a 2 gm sample requires about one hour. Isotopic contamination of the UF_6 is avoided by confining the use of a reactor tube to a small range of isotopic concentrations and by a thorough washing of the reactor tube between the fluorination of samples of different isotopic concentration. (auth)

8420 K-215

Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.

CONTINUOUS METHOD OF REDUCING UF_6 TO UF_4 BY MEANS OF TRICHLOROETHYLENE. A. L. Allen, H. A. Bernhardt, W. L. Maroney, B. J. Massey, J. G. Schaffner, and E. A. Wiggin. May 17, 1948. Decl. Feb. 25, 1957. 19p. Contract [W-7405-eng-26]. $0.30(OTS).

Procedures are presented for the continuous reduction of UF_6 to UF_4 by gaseous trichloroethylene on a laboratory scale. Optimum flow rates and temperatures are given along with other operational data. A brief history of the reactor development is also included. (auth)

8421 K-240

Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.

METHODS FOR THE PREPARATION OF CUBES CONTAINING URANIUM COMPOUNDS AND THE RECOVERY OF THE URANIUM SALTS. W. L. Maroney, B. J. Massey, J. G. Schaffner, E. Staple, E. A. Wiggin, A. D. Callihan, and S. Visner. July 7, 1948. Decl. Feb. 25, 1957. 35p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

A satisfactory method for the production of cubes containing U and F in the atomic ratio of 1 to 6 has been developed. It was necessary that these cubes be chemically stable, uniform in chemical composition, resistant to mechanical shock, and have strict dimensional tolerances. Procedures are described for the preparation of suitable U salts which could be molded into cubes by using fluorocarbon polymers as binders. Finally, procedures are given for the recovery of the U compounds from various waste products after experiments with the cubes had been completed. (auth)

8422 K-429

Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.

THE HIGH TEMPERATURE HYDROGENATION OF URANYL AMMONIUM PHOSPHATE. H. A. Bernhardt, W. Davis, Jr., J. R. Flannary, and R. J. Heus. July 15, 1949. Decl. Mar. 4, 1957. 25p. (KLO-121). $4.80(ph OTS); $2.70(mf OTS).

Conversion of uranyl ammonium phosphate (UAP) to UF_4 can be accomplished by reduction of the former at 500 to 700°C followed by hydrofluorination of the reduced material. The first process, reduction, has been studied in the temperature range 500 to 700°C and the results are reported. At 500°C, reduction of samples weighing about 8 g is incomplete at a H_2 flow rate through the bed of 10 to 18 ml/min at STP. At 700°C nearly complete reduction has been observed at reaction times of about one hour. On the assumption that the data can be treated kinetically it has been found that the over-all reduction reaction follows a first-order rate law within experimental error. An apparent activation energy for reduction of 26 k cal/mole has been estimated. Reduction of hexavalent U in a N atmosphere has been found to occur above a temperature of 500°C. This is consistent with thermal dissociation of ammonia from the starting material to N and H and reduction of the U by the H thus formed. (auth)

8423 K-548

Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.

FLUORINATION OF URANIUM PILE SLUGS WITH ELEMENTAL FLUORINE. L. A. Barry, G. H. Montillon, and R. Van Winkle. Dec. 30, 1949. Decl. Feb. 23, 1957. 35p. $6.30(ph OTS); $3.00(mf OTS).

An equation was derived which will predict rate of U consumption during fluorination of metal slugs, with an accuracy of ~2 to 3%. This relationship requires the measurement of only the F_2 flow and the exit gas stream temperature. At some metal temperature above 1600°F, a phase change occurs in the reaction products giving rise to uncontrollable reaction conditions. Based on this experience with cold slugs, recommendations were made for the construction and operation of equipment for the fluorination of irradiated U slugs. (auth)

8424 K-882

Massachusetts Inst. of Tech., Oak Ridge, Tenn. Engineering Practice School.

RECOVERY OF URANIUM FROM URANIUM–ZIRCONIUM ALLOYS. M. L. Baker, P. M. Lang, and W. S. Scheib. Mar. 4, 1952. Decl. Feb. 23, 1957 31p. For [Oak Ridge National Lab., Contract W-7405-Eng-26]. Subcontract 70. $0.35(OTS).

The investigation included: (1) studies of the chemical characteristics of the dissolution of Zr–U alloy in concentrated sulfuric acid, (2) studies of the reaction rate of the alloy with H_2SO_4 as a function of the concentration, (3) separation of the dissolved salts by anion exchange, and (4) leaching of the hydroxides, sulfates, and oxides of U from those of Zr by boiling concentrated HNO_3. The following major conclusions were drawn from the investigation: (1) U–Zr alloys can be dissolved in concentrated H_2SO_4. (2) The reaction rate reaches a sharp maximum at an acid concentration of 30.4 N. Further work to evaluate the effectiveness of ion exchange as a means of separation of the metals is recommended. (auth)

8425 K-1137

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

FLUORINATION OF URANIUM OXIDES WITH BROMINE TRIFLUORIDE FOR MASS SPECTROMETER ASSAYING. R. E. Greene and G. S. Petit. Aug. 25, 1954. Decl. Feb. 23, 1957. 15p. Contract W-7405-eng-26. $0.30(OTS).

Bromine trifluoride has been successfully employed for the conversions of UO_2 to UF_6 for assaying by the mass spectrometer. BrF_3 is allowed to react with UO_2 in a bomb type reactor, from which the reaction products, Br and UF_6, are vacuum distilled into a special type of cold trap. The Br is removed by a low temperature fractional distillation, leaving UF_6 or sufficient purity for introduction into the mass spectrometer. The method has proved to be rapid, economical, reliable, and readily adaptable to routine use. (auth)

8426 K-1184

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

EVALUATION OF BROMINE TRIFLUORIDE AS A REAGENT FOR THE ROUTINE DETERMINATION OF URANIUM. J. F. Steinbach. Feb. 9, 1955. Decl. Mar. 13, 1957. 16p. Contract W-7405-eng-26. $0.25(OTS).

The analytical distillation of uranium hexafluoride from samples treated with bromine trifluoride has been evaluated as a routine separation method. As compared to solvent extraction now used, the method investigated is tedious, time-consuming and for reasonable operation times is not specific. Analytical recovery of the uranium is difficult without additional separation of impurities distilled with the uranium hexafluoride. Recovery of uranium from insoluble materials is low. (auth)

8427 K-1201

Carbide and Carbon Chemicals Co. K-25 Plant, Oak Ridge, Tenn.

U^{235} CONTENT OF NATURAL URANIUM. R. E. Greene, C. A. Kienberger, and A. B. Meservey. June 9, 1955. 15p. Decl. Feb. 25, 1957. 15p. Contract W-7405-eng-26. $0.30(OTS).

The high precision determination of the U^{235} content of natural U has been repeated using improved techniques and new materials. The new value is 0.71057 ± 0.00059 wt. %, as compared with the former value of 0.7115 ± 0.0014 wt. % of this isotope. (auth)

8428 M-3022

Du Pont de Nemours (E. I.) & Co. Jackson Lab., Wilmington, Del.

TENTATIVE DYE WORKS PROCESS FOR THE MANUFACTURE OF C-106 (PURIFIED). C. W. Maynard. 7p. May 17, 1943. Decl. Mar. 6, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The purification of uranyl nitrate hexahydrate, UNH, by the ether-water extractions of the crude UNH product is described. (C.W.H.)

8429 Y-522

Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.

THE REACTION OF ANHYDROUS HYDROGEN FLUORIDE WITH VARIOUS PHOSPHATES OF URANIUM. D. W. Kuhn. Nov. 17, 1949. Decl. Mar. 9, 1957. 18p. Contract W-7405-Eng-26. $3.30(ph OTS); $2.40(mf OTS).

The reactions of anhydrous hydrogen fluoride with uranyl ammonium phosphate, uranyl hydrogen phosphate, and uranous oxypyrophosphate at 400 to 400°C have been studied with the purpose of identifying the volatile reaction products. It has been found that in such hydrofluorination reactions the P is evolved as either PF_5 or HPF_6, and can be collected as an HPF_6 solution by cooling the effluent gases. In addition, several experiments have been performed which were designed to show how much hydrogen fluoride can be recovered from the product gases for recycle. Data from distillation experiments for recovery of hydrogen fluoride are included in this report. (auth)

Refer also to abstracts 8433, 8486, and 8514.

WASTE DISPOSAL

8430 A-3789(Sect. 8)

Du Pont de Nemours (E. I.) & Co. TNX Div., Wilmington, Del.

CLINTON LABORATORIES PROCESS MANUAL. SECTION 8. WASTE DISPOSAL—WET B PROCESS. W. E. Kirst. [1943]. Decl. Feb. 12, 1957. 21p. $4.80(ph OTS); $2.70 (mf OTS).

All waste solutions and slurries containing varying amounts of radioactive materials which require permanent storage are neutralized to a pH of 7 to 7.5 by the addition of caustic and held indefinitely in underground concrete tanks. If the contents are not too greatly contaminated by radioactive material, they are discharged through the retention ponds to the river. If sufficiently contaminated, they are diverted to the underground tanks for permanent storage. All cooling water is collected in one of two retention ponds and analyzed for radioactivity. If satisfactory, it is discharged to the river. By proper control, radioactive materials in these ponds should never reach a concentration where their discharge to the river is not possible, although it may be necessary infrequently to choose an optimum time, or discharge in portions. Storage facilities are described in detail. (auth)

8431 CF-51-12-50

Oak Ridge National Lab., Tenn.

ABSORPTION OF WASTE HYDROGEN FLUORIDE. V. J. Reilly. Dec. 7, 1951. Decl. Feb. 15 1957. 10p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Experiments on four methods of disposal of HF from the Pu peroxide hydrofluorination furnace in a waste gas stream are discussed, namely: (1) condensation as a liquid for reuse or disposal, (2) venting the filtered gas to the stack, (3) absorption in a lime slurry, and (4) absorption in a limestone bed. Although no definite choice of one method to the exclusion of the others was made it was felt that venting to the stack merited serious consideration. (L.M.T.)

8432 CF-56-1-131

Oak Ridge National Lab., Tenn.

PACKAGING OF RADIOACTIVE WASTE IN CERAMIC BODIES. Lane Mitchell. Jan. 30, 1956. 24p. Contract [W-7405-Eng-26]. $4.80(ph OTS); $2.70(mf OTS).

It has been suggested that waste disposal problems might be solved by packaging the concentrated activity in relatively small packages which are not readily leached and burying them in deep pits. With this purpose, a preliminary investigation was made of the adsorption and absorption of radioisotopes by porous ceramic slugs. Procedures for the tests are outlined, and results are summarized. (B.J.H.)

8433 KAPL-117

Knolls Atomic Power Lab., Schenectady, N. Y.

LIQUID WASTE DISPOSAL. J. A. Ayres, G. E. McCullough, and E. C. Pitzer. Jan. 24, 1949. Decl. Mar. 12, 1957. 79p. Contract W-31-109-eng-52. $12.30(ph OTS); $4.50 (mf OTS).

Data are presented on waste disposal work accomplished at KAPL up to the design of equipment for installation in the Separations Process Research Unit. (auth)

8434 KLX-1302

Kellex Corp., New York.

PROGRESS REPORT [FOR] PERIOD APRIL 1–30, 1950. Job 24-A. May 5, 1950. Decl. Mar. 2, 1957. 38p. Contract AT(30-1)-850. $6.30(ph OTS); $3.00(mf OTS).

Studies are summarized on the evaluation of the effectiveness of particle conditioning techniques with a view toward obtaining high separation efficiencies in various types of gas cleaning equipment, and the development of an efficient method of decontaminating crib wastes using ion exchange resins. (L.M.T.)

8435 MLM-652

Mound Lab., Miamisburg, Ohio.

REPORT FOR LIQUID WASTE DISPOSAL RESEARCH–SEPTEMBER 3, 1951 TO DECEMBER 3, 1951. Jan. 7, 1952. Decl. Feb. 7, 1957. 19p. Contract AT-33-1-GEN-53. $3.30(ph OTS); $2.40(mf OTS).

The difference in decontamination obtained with TBP Metal Recovery Wastes and fresh, evaporated and neutralized. Purex wastes on repeated scavenging with ferrous sulfide and ferrous hydroxide-calcium phosphate precipitations has been traced to the insufficient amount of phosphate in the Purex waste to provide an excess over the stoichiometric amount equivalent to the Ca added. The Purex waste from treating three- to five-year aged slugs responds

to scavenging treatment similarly to the fresher wastes, providing an oxidizing treatment to break up Ru complexes is employed. The removal of Cs by ferrous sulfide is not appreciably affected by pH. Better removal is obtained when the precipitate is formed by addition of solid Na sulfide rather than by addition of a solution of Na sulfide. Fluoride interferes with removal. Decontamination factors of the order of 500 were obtained under optimum conditions such as an Fe concentration of 0.02 molar and inactive Cs concentration of about 10^{-8} molar in 30% Na nitrate solution. (auth)

ENGINEERING

HEAT TRANSFER AND FLUID FLOW

8436 CF-55-9-107
Oak Ridge National Lab., Tenn.
STUDIES OF CORE FLOW BY ELECTROLYTE INJECTION AND BY DYE INJECTION. J. M. Trummel and E. C. Lindly. Sept. 22, 1955. Decl. Feb. 16, 1957. 27p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Two experimental methods were used to obtain information on the flow of water through full-scale models to investigate the characteristics of flow such as the diffusivity and distribution of residence times in the zone of internal heating. (F.S.)

8437 CF-56-3-153
Oak Ridge National Lab., Tenn.
THE RHEOLOGICAL BEHAVIOR OF THORIUM OXIDE SLURRIES IN LAMINAR FLOW. P. R. Crowley and A. S. Kitzes. Mar. 19, 1956. 24p. Contract [W-7405-Eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The flow behavior of aqueous thorium oxide slurries is an integral part of the information needed in designing the blanket system of a thorium oxide slurry breeder reactor. As a result of an investigation designed to determine these rheological properties, it was found that thorium oxide slurries are plastic non-Newtonian fluids at ambient temperatures. It was found that the viscosity can be expressed in terms of shear stress, τ, and concentration, S, by the equation $\mu_S = \mu_L\ [1.2 + S(c\tau^{-0.2} + 1)^3/1.2 - 2S(c\tau^{-0.2} + 1)^3]$. The laminar flow of thorium oxide slurries in circular conduits is described by the equation $8V/g_c D = 4/\mu_L \tau_D{}^3 \int_{\tau_y}^{\tau_D} \tau^3 [1.2 - 2S(c\tau^{-0.2} + 1)^3/1.2 + S(c\tau^{-0.2} + 1)^3]\, d\tau$, and the yield stress is related to concentration by the equation $\tau_y = [c/(1.2/2S)^{1/3} - 1]^5$. The constant c in the above expressions is unique for a given slurry (independent of concentration) and may be determined from any one viscosity, flow rate, or yield stress of that slurry. (auth)

8438 LRL-14
California Research and Development Co., Berkeley, Calif.
SUMMARY OF HEAT CONDUCTION EQUATIONS FOR SLABS, RODS, SPHERES, AND TUBES. STEADY STATE, TRANSIENT, AND PERIODIC STEADY STATE WITH UNIFORM HEAT GENERATION WITHIN THE SOLID. N. C. Ostrander. Mar. 29, 1951. Decl. Feb. 26, 1957. 19p. ([LWS-2232]). $3.30(ph OTS); $2.40(mf OTS).

The steady state, transient, and periodic steady state equations which have been developed for cases of uniform heat generation with a film coefficient at the boundary or boundaries are summarized. The geometrical shapes considered are those simple shapes which are potential components of the Mark II target. The equations have been developed primarily for regular pulsed wave PW operation, however, the steady state solutions (for continuous wave operation) are included, and the means of extending the equations to cover cases of irregular pulsing, such as may occur during sparking in the accelerator, is demonstrated. (auth)

8439 LWS-24525
California Research and Development Co., Livermore, Calif.
FREE CONVECTIVE COOLING OF PARALLEL VERTICAL PLATES. J. C. Ekvall. Aug. 28, 1952. Decl. Feb. 27, 1957. 15p. (CRD-T1-154). $3.30(ph OTS); $2.40(mf OTS)

A method is presented for analyzing the problem of free convective cooling of parallel plates by vertical flow of a fluid between them. The temperature of the fluid was assumed to vary linearly with the height of the plates; so this method applies only to plates considerably longer than 16 inches. As an example of the use of the method, free convective cooling of the flat head of the cylindrical vacuum vessel was analyzed. (auth)

Refer also to abstract 8439.

MATERIALS TESTING

8440 A-3789(Sect. 13)(Del.)
Du Pont de Nemours (E. I.) & Co. TNX Div., Wilmington, Del.
CLINTON LABORATORIES PROCESS MANUAL. SECTION 13. MATERIALS OF CONSTRUCTION. Mar. 31, 1944. Decl. with deletions Feb. 12, 1957. 29p. $4.80(ph OTS); $2.70(mf OTS).

Metals, alloys, gaskets, electrical insulation, and glass were irradiation and corrosion tested for possible use in the Lanthanum Fluoride Process. (D.E.B.)

WASTE DISPOSAL

8441 KLX-1067
Kellex Corp., New York.
EFFLUENT CONCENTRATION. Report for Period July 18, 1949–March 7, 1950. Mar. 7, 1950. Decl. Feb. 25, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Test runs of an evaporator on waste from Hanford metal recovery process are described. (T.R.H.)

MINERALOGY, METALLURGY, AND CERAMICS

CERAMICS AND REFRACTORIES

8442 CT-3491
Massachusetts Inst. of Tech., Cambridge.
SUPPLEMENTARY REPORT ON THE PREPARATION OF LARGE MAGNESIA CRUCIBLES. Supplement to Report CT-3373. F. Cunningham, K. Lundell, and A. Zais. Apr. 24, 1946. Decl. Feb. 18, 1957. 21p. Contract W-7405-eng-175. $4.80(ph OTS); $2.70(mf OTS).

Modifications in techniques for the preparation of magnesia for crucible production, crucible fabrication, and firing of crucibles are described. (cf CT-3373.) (C.W.H.)

CORROSION

8443 BMI-998
Battelle Memorial Inst., Columbus, Ohio.
APPARATUS FOR VISUAL STUDY OF CORROSION BY HOT WATER. Daniel R. Grieser and Eugene M. Simons. May 4, 1955. Decl. Feb. 14, 1957. 20p. Contract W-7405-eng-92. $0.30(OTS).

The design and operation of a windowed autoclave suit-

able for studying corrosion of materials by pressurized hot water is described. The technique for taking time-lapse motion pictures of a specimen from the instant of contact with the high-temperature water until the corrosion is complete is explained. A typical set of pictures is presented, showing the progress of the attack. Pressure and temperature measurements provide a means for rough computation of corrosion rates. The results of several tests of unclad and zirconium-clad unbonded uranium–zirconium alloy cores subjected to pressurized hot water are presented. (auth)

8444 BNL-179
Brookhaven National Lab., Upton, N. Y.
STATIC CORROSION BEHAVIOR OF CONSTRUCTION MATERIALS IN AN ENVIRONMENT OF LIQUID BISMUTH BASE METALS AT 550°C. M. A. Cordovi. Mar. 1, 1952. Decl. Feb. 18, 1957. 40p. $0.35(OTS).

This report presents preliminary results obtained in a static corrosion test program which was designed to screen promising construction materials for a proposed liquid fuel power breeder reactor. Iron, C steel, a graded series of ferritic chromium–molybdenum steels, and several stainless steels have been exposed for periods of 500 to 2000 hr to the action of Bi, Bi–Pb eutectic and Bi–Pb–Sn eutectic, with and without uranium (0.4% to 2% U) additions. All the experiments were conducted at 550°C, but one series of tests in U–Bi was also carried out at 650°C. A novel type of corrosion sample was used in this investigation to determine quantitatively the extent of both surface and subsurface attack. The accuracy of this method notwithstanding, the results obtained to date have no design significance and serve only to indicate general trends. Statistically proper evaluation of the results cited herein is impossible because the majority of the data were derived from single tests Furthermore, the influence on corrosion rate of what are believed to be important incidental factors, internal or external to the metal, has not been given serious consideration in this preliminary phase of the corrosion program. The apparent trends exhibited by the various materials tested are summarized. On the basis of present information, it would seem that the steel and liquid metal combinations which show the greatest promise of success under the required temperature conditions are the $2\frac{1}{4}$ Cr–1 Mo and 5 Cr–$\frac{1}{2}$ Mo–1.5 Si steels in U–Bi and U–Bi–Pb (eutectic). (auth)

8445 BNL-2439
Brookhaven National Lab., Upton, N. Y.
LMFR PROGRESS LETTER FOR JUNE 1955. F. T. Miles. July 25, 1955. Decl. Feb. 18, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

8446 BNL-2538
Brookhaven National Lab., Upton, N. Y.
LMFR PROGRESS LETTER FOR SEPTEMBER 1955. F. T. Miles. Oct. 27, 1955. Decl. Feb. 18, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

8447 CE-2544
[Chicago. Univ. Metallurgical Lab.]
SINGLE SLUG ELECTRICALLY INTERNAL HEATED CORROSION UNIT. Final Report on a Phase of Problem No. 124-ML-51-17. R. N. Lyon. June 1, 1944. Decl. Feb. 20, 1957. 5p. Contract W-7401-eng-37. $1.80(ph OTS); $1.80(mf OTS).

A method for internally heating Hanford slugs with bonded jackets for corrosion studies of the jackets and bonds was designed. The equipment developed was tested and found entirely satisfactory. It is recommended that the equipment be adapted for use at ML and CMX for the purpose of testing slugs, jackets, and bonds under internal heating conditions. (auth)

8448 CF-56-2-72
Oak Ridge National Lab., Tenn.
SUMMARY OF TOROID RUN NO. 81: TYPE 347 STAINLESS STEEL, TITANIUM 75A AND ZIRCALOY-2 IN FERNALD AND D-17 THORIUM OXIDE SLURRIES AT 300°C AND 26 fps. EFFECT OF FLUORIDE. G. E. Moore and E. L. Compere. Feb. 14, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8449 CF-56-11-72
Oak Ridge National Lab., Tenn.
CORROSION BY AQUEOUS REACTOR FUEL SOLUTIONS. H. F. McDuffie. Nov. 14, 1956. Decl. Mar. 12, 1957. 58p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60(mf OTS).

A review of experimental work on corrosion of reactor systems by aqueous reactor fuel solutions is presented. Corrosion of Ti, Pt, austenitic stainless steels, Zr, Ni alloys, and Zr alloys by UO_2SO_4 solutions is discussed. (T.R.H.)

8450 CT-687
[Chicago. Univ.] Metallurgical Lab.
COATINGS AND CORROSION. PART III OF REPORT FOR MONTH ENDING MAY 29, 1943. J. P. Howe. Decl. Feb. 18, 1957. 33p. Contract [W-7401-eng-37]. (A-867). $7.80(ph OTS); $3.40(mf OTS).

8451 CT-1440
Clinton Labs., Oak Ridge, Tenn.
CORROSION AND ALUMINUM ACTIVITY IN PILE WATER. Progress Report—Problem Assignment Nos. 124-X1E and 124-X11E. W. P. Jesse, M. C. Leverett, W. D. B. Spatz, and R B. Briggs. May 15, 1944. Decl. Feb. 18, 1957. 32p. Contract [W-7405-eng-39]. (A-2283). $6.30(ph OTS); $3.00(mf OTS).

The corrosion rate of 2S aluminum in the X pile under simulated W conditions is about $\frac{1}{7}$ that calculated from the Al^{28} activity in the pile effluent water. It is proposed that the discrepancy is due to the aluminum oxide film where, it is postulated, Al^{28} atoms are formed and from which they escape into the cooling water by diffusion. The corrosion rate measured by weight loss of the slugs is normal (about 0.001 inch/month) for the rather corrosive conditions used. No effect of pile radiation on corrosion rate was observed. The corrosion rate was slightly less at 92°C than at 77°C. A possible empirical relationship between Al^{28} activity in the water and corrosion rate is mentioned. (auth)

8452 CT-3035
[Chicago. Univ. Metallurgical Lab.]
AQUEOUS CORROSION OF TUBALLOY[U]-SILICON ALLOYS. Joyce M. Hopkins, W. W. Binger, and Frederick Nelson. June 14, 1945. Decl. Feb. 19, 1957. 26p. Contract W-7401-eng-37. $4.80(ph OTS); $2.70(mf OTS).

Results are presented from studies on the corrosion of heat-treated cast U–Si alloys in relatively pure, air-free water. The primary purpose of the investigation was to provide information needed to evaluate these alloys as possible slug material. (C.H.)

8453 HW-12964
[Hanford Works, Richland, Wash.]
CORROSION TESTS ON CARBOLOY IN REDOX PROCESS SOLUTIONS. W. W. Koenig. Apr. 6, 1949. Decl. Feb. 20, 1957. 8p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Static, total immersion corrosion tests were carried out at room temperature with Carboloy of various grades in Redox Process solutions. Results indicate that none of the Carboloy samples used are satisfactory for use with Redox Process solutions. (B.J.H.)

8454 HW-17159
Hanford Works, Richland, Wash.
CORROSION OF STELLITE IN MJ-1 STREAMS. W. W. Koenig. Mar. 8, 1950. Decl. Feb. 19, 1957. 11p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Corrosion testing of stellite–stainless steel cylinders in Redox Process streams was completed. Complete results are tabulated. (B.J.H.)

8455 HW-21160
[Hanford Works, Richland, Wash.]
CORROSION DATA FOR TYPE 347 STAINLESS STEEL PIPE WASTE METAL RECOVERY PROCESS. Kenneth L. Sanborn. May 22, 1951. Decl. Mar. 2, 1957. 7p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

8456 HW-27434
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
CORROSIVE EFFECTS OF HEAVY METAL IONS ON ALUMINUM. G. S. Fujioka. Mar. 17, 1953. Decl. Feb. 20, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

The corrosive effects of reactor cooling water which contained traces of heavy metal ions on aluminum were investigated. Results indicated that Cu impurities contributed to the pitting corrosion of Al. (C.W.H.)

8457 IDO-14035
American Cyanamid Co. Atomic Energy Div., Idaho Falls, Idaho.
CORROSION OF METALS AND ALLOYS IN SIR AND STR PROCESS STREAMS. R. H. Perkins and C. M. Slansky. Mar. 1952. Decl. Feb. 23, 1957. 148p. Contract AT(10-1)-177. $22.80(ph OTS); $7.20(mf OTS).

The available classified and unclassified literature is reviewed relative to the corrosion of metals and alloys in solutions similar to SIR and STR fuel-recovery process streams. Illium R, Carpenter 20, and Hastelloy F are the likely materials of construction for the SIR dissolver while Type 347 or 304 E.L.C. stainless steel is suitable for handling process streams following the dissolver. Corrosion in the STR dissolver is largely dependent on the presence or absence of an oxidizing agent. In the absence of a strong oxidizing agent, the dissolver can be made of monel, Ag, or Hastelloy F for temperatures up to boiling. Pt was the only metal reported that was resistant to mixtures of HF and strong oxidizing agents. The processing equipment following the STR dissolver can be made of either Type 347 or 304 E.L.C. stainless steel. (auth)

8458 IDO-14213(Del.)
[American Cyanamid Co. Atomic Energy Div., Idaho Falls, Idaho.]
CORROSION OF ALUMINUM–URANIUM FUEL ASSEMBLIES IN SF STORAGE BASIN WATER. Supplementary Report No. 1. W. J. Guay. Feb. 6, 1953. Decl. with deletions Feb. 8, 1957. 3p. Contract [AT(10-1)-177]. $1.80 (ph OTS); $1.80(mf OTS).

8459 ORNL-195
Oak Ridge National Lab., Tenn.
THE CORROSION OF STAINLESS STEELS IN ALUMINUM NITRATE–SODIUM DICHROMATE–URANYL NITRATE SOLUTION. A. R. Olsen. Nov. 3, 1949. Decl. Mar. 19, 1957. 8p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

Various types of welded stainless steel specimens were exposed to a solution simulating in composition that present in a solvent extraction process in the 1200 Area at Oak Ridge National Laboratory. The specimens were exposed singly and in contact with other stainless steels in an attempt to determine the formation rate of corrosion products on or adjacent to welded areas. Stainless steel types 302, 304, 316, and 347 were the materials under consideration for the extraction columns. The results of the tests indicated that none of the stainless steels were subject to serious corrosion attack. Types 302 and 347 showed slightly superior corrosion resistance which was particularly evident in contact couples of these two materials. (auth)

8460 ORNL-298
Oak Ridge National Lab., Tenn.
THE CORROSION OF BERYLLIUM IN SIMULATED COOLING WATER FOR THE PROPOSED DEVELOPMENT REACTOR. Interim Report [for] Period September 1946 to December 1948. James L. English. Mar. 17, 1949. Decl. Feb. 7, 1957. 129p. Contract W-7405-eng-26. $19.80(ph OTS); $6.30(mf OTS).

Corrosion studies on vacuum cast and extruded Be in simulated cooling water depend on metal purity, metallurgical treatment of metal, quality of cooling water, and the effect of solution velocity and stagnant areas in the water. The presence of O_2 tended to reduce corrosion for Be but increased it in bimetallic couples. (F.S.)

METALS AND METALLURGY

8461 ANL-4582
Argonne National Lab., Lemont, Ill.
EFFECTS OF GALLIUM ON MATERIALS AT ELEVATED TEMPERATURES. Supplementary Report. W. D. Wilkinson. Aug. 1953. Decl. Feb. 13, 1957. 33p. Contract W-31-109-eng-38. $0.40(OTS).

The reactions and solubility of U in Ga at elevated temperatures have been studied by chemical and x-ray analysis of the reaction products and by metallographic procedures. In addition, tests have been made to determine the behavior in Ga of U hydride, UO_2, and of the product between U and Ga. (auth)

8462 ANL-5439(Del.)
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT [FOR] JANUARY, FEBRUARY, AND MARCH 1955. Sept. 1955. Decl. with deletions Feb. 12, 1957. 59p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

Fabrication of Zircaloy-clad U alloy fuel blates, welded end closures for EBWR fuel plates, and fabrication of U–Zr fuel elements for EBR-II are reported. Melting and casting of high-temperature U alloys, and the investigation of a satisfactory wash for graphite crucibles were studied. Specimens of ThO_2–U_3O_8 ceramic fuel bodies were fabricated by wet milling, dewatering, and dry pressing. Work was continued on the preparation of α–U single crystals, thermal cycling of U jacketed in Ta, expansion coefficients of Pu–U alloys, solubility of Zr in β–U, and the equilibrium phase diagram of Nb–U–Zr alloys. High temperature aqueous corrosion of Al and the effects of Fe and Si additions on the corrosion resistance of Al–Ni alloys were investigated. The mechanism of Zr corrosion, the corrosion resistance of U alloys, the effect of fabrication variables on the behavior of U under irradiation, the effect of irradiation on Pu–U alloys and on uranium oxide–zirconium cermets are given. (J.E.D.)

8463 BMI-69
Battelle Memorial Inst., Columbus, Ohio.
JACKETING OF URANIUM FOR HIGH-TEMPERATURE SERVICE. II. BARRIER COAT MATERIALS. H. A. Saller and J. T. Stacy. June 1, 1951. Decl. Feb. 13, 1957. 32p. Contract W-7405-eng-92. $0.35(OTS).

The effectiveness of various barrier metals between U,

U-Mo alloys, or U-Nb alloys and a low-C steel was measured by sandwich-type diffusion-to-failure tests. The efficacy of the barrier coat was judged on the basis of the time and temperature required for failure vs. the thickness of the barrier-coat metal. Most of these evaluations were made between 900 and 1100°C over times ranging up to at least 100 hr at tmperature. The following metals were tested against pure U: Ag, Ta, W, Mo, Nb, Va, Zr, Ti, Be, and Pa. The diffusion pack used in the procedure is described. Photos show the effect of the tests on the various metals. (auth)

8464 BMI-765
Battelle Memorial Inst., Columbus, Ohio.
AN APPLICATION OF THE CONCEPTS OF PARTICLE PACKING TO THE CONSOLIDATION OF SILICON CARBIDE POWDERS. C. G. Harman, J. Shinn, Jr., and H. E. Wagner. Aug. 15, 1952. Decl. Feb. 18, 1957. 30p. Contract W-7405-eng-92. $0.35(OTS).

The basic concepts and published information on the packing of particles of 1, 2, and 3 sizes are reviewed. Experimental work was conducted to check the applicability of reported results to the immediate problem of the fabrication of thin forms of silicon carbide and to establish suitable sizes and size distributions. Studies were made of the packing of silicon carbide powders within the size range of 210 microns to minus 0.5 micron. The results indicated that for closely sized fractions of less than 48 microns, packed porosities increased with decreasing particle size. In general, the finer the particles, the stronger were the pressed forms. The limited study of ternary packing indicated a minimum void composition (26.2% of porosity when pressed at 20,000 psi) for samples containing: coarse (177 to 149 microns), 65%; intermediate (25 to 10 microns), 17.5%; and 17.5% of minus 4 microns. (auth)

8465 BMI-924
Battelle Memorial Inst., Columbus, Ohio.
NONDESTRUCTIVE MEASUREMENTS OF ALUMINUM AND NICKEL LAYERS ON URANIUM. E. W. Glaze, M. L. Rhoten, B. A. Kulp, K. D. Cooley, and S. A. Wenk. July 12, 1954. Decl. Feb. 14, 1957. 34p. Contract W-7405-Eng-92. $0.30(OTS).

Two approaches to the problem of nondestructively measuring aluminum and nickel layers on uranium have been investigated. One utilizes eddy currents, the other x rays. Both methods have yielded results which are encouraging. Probes with sufficient sensitivity to make quantitative thickness measurements of nickel electroplated on flat-plate uranium have been designed and constructed for use with the Multitest eddy-current instrument. Analyses of test results indicate that the plating thickness can probably be measured with an accuracy of ±50 μ in the 0.3-mil range. On aluminum-clad flat-plate uranium fuel elements, cladding thicknesses from 5 to 35 mils have been measured with the Multitest instrument. With adequate calibration, it should be possible to determine the thickness of the cladding to an accuracy of at least ±0.5 mil. An experimental apparatus was designed and built for an x-ray method of measuring the thickness of electroplated nickel on flat-plate uranium. This device allows changing the angles of both the sample and the Geiger tube with respect to the incident x-ray beam. It was found that the angles were not important, as long as the same angles were used for one complete set of readings. Adjustable tantalum slits provide for variation in the size of the area being examined. With an accuracy of 2 per cent inherent in the x-ray intensity meter, it was possible to measure 10 per cent variation in coating thickness. Thus, at 0.3 mil it is possible to measure variations of 30 μ in. (auth)

8466 BMI-934
Battelle Memorial Inst., Columbus, Ohio.
THE ROLL CLADDING OF URANIUM WITH ALUMINUM. H. A. Saller, S. J. Paprocki, and J. F. Delaney. Aug. 3, 1954. Decl. Feb. 14, 1957. 30p. Contract W-7405-Eng-92. $0.30(OTS).

The possibility of producing flat-plate aluminum-clad uranium fuel elements by rolling was investigated. The aluminum-to-uranium bond was found to be weak because of the formation of intermetallic compounds and the oxidation of the core during the preheat. Several interface coats on the uranium (chromium, copper, nickel, and zinc electroplates, as well as an Al–Si* dip coat) were tried. Nickel and copper gave the best bonds; the other materials gave relatively weak bonds. Nickel was selected for concentrated development on the basis of its apparent superior corrosion resistance when in contact with uranium. Factors affecting the quality of the nickel electroplate were investigated, and the anodic treatment was found to be critical. Heat treatment also has considerable effect on the bond; the bonds weaken after 5 min at 900 to 1100°F. An extruded aluminum sheath will be used for cladding the fuel plates, and aluminum end plugs will be inserted in each end. The best bonds were obtained with copper-plated end plugs hot pressed to the ends of the plates. Edge bonding was obtained by rolling through grooved rolls. It was concluded that roll cladding can be used as a method of producing the aluminum-clad fuel plates. (auth)

8467 BMI-959
Battelle Memorial Inst., Columbus, Ohio.
THE CASTING OF RADIATOR-TYPE FUEL ELEMENTS OF THE URANIUM–CHROMIUM EUTECTIC ALLOY. H. A. Saller, R. F. Dickerson, A. W. Hare, and W. E. Murr. Oct. 22, 1954. Decl. Feb. 13, 1957. 22p. Contract W-7405-eng-92. $0.25(OTS).

The feasibility of casting of a bare, radiator-type fuel element of the U–Cr eutectic alloy was investigated. Four 4-in.-high castings of the radiator-type element were attempted. The best casting was made in a sectioned graphite mold which had been heated to 1550°F. The alloy was poured at 1900°F. The core materials for the graphite mold were investigated on a small-scale basis. The core materials that appeared to be satisfactory were then used in the 4-in.-high radiator-type castings. Type 302 stainless steel rods sprayed with a 0.010-in.-thick coating of MgO proved to be the best core material. Three grooved plates, longitudinal sections of the bare, radiator-type fuel element, were cast in both a preheated graphite mold and graphite molds that were at room temperature. A successful grooved plate was cast in a graphite mold which was preheated to 1400°F. The alloy was poured at 1900°F. (auth)

8468 BMI-965
Battelle Memorial Inst., Columbus, Ohio.
CLASSIFICATION OF STRINGERS AND IDENTIFICATION OF INCLUSIONS IN REACTOR-GRADE URANIUM. H. A. Saller, ed. Nov. 29, 1954. Decl. Feb. 26, 1957. 22p. Contract W-7405-eng-92. $0.30(OTS).

A system of classification based on identification of the types of stringers and inclusions found in the current reactor-grade of U metal is presented. Metallographic techniques for the preparation and examination of U are described. The positive identification of various inclusions was accomplished by two x-ray techniques. Typical examples of inclusions are shown. (J.A.G.)

8469 BMI-983
Battelle Memorial Inst., Columbus, Ohio.
THE DEVELOPMENT OF CAST FUEL ELEMENTS. H. A.

Saller, R. F. Dickerson, A. W. Hare, and N. E. Daniel. Feb. 21, 1955. Decl. Mar. 1, 1957. 26p. Supported by Atomic Power Development Associates through Contract W-7405-eng-92. $0.30.

The feasibility of casting a bare, radiator-type fuel element of the uranium–chromium eutectic alloy (5 w/o chromium) was investigated. Castings of both perforated-hexagonal subsections and grooved-plate longitudinal sections of the perforated-hexagon subsection of the fuel element were attempted. Eight successful perforated-hexagon castings were made. Five were 2 in. high and three were 4 in. high. The castings were made in sectional graphite molds which had been heated to 1600 F. The eutectic alloy was poured at 2000 F. The cores for the castings were Type 302 stainless steel rods with a 0.020-in.-thick coating of MgO applied by electrophoresis. A 24-in.-long grooved-plate subsection was gravity cast into a preheated split-type graphite mold. Gravity castings of 4-in.-long grooved-plate subsections were cast in ceramic molds which were made by the lost-wax process. Two 12-in.-long castings were centrifugally cast in a preheated and a cold steel mold, respectively. Because of the high thermal stresses from the drastic quench of the hot metal by the relatively cold molds, the castings cracked. The joining of uranium–chromium eutectic alloy specimens by diffusion was given preliminary study. Metal powders and various electroplated metals were tried as bonding agents at the interface to be joined. (auth)

8470 BMI-984

Battelle Memorial Inst., Columbus, Ohio.

THE PROPERTIES OF HIGH-URANIUM ALLOYS CONTAINING ZIRCONIUM OR CHROMIUM. H. A. Saller and F. A. Rough. Feb. 21, 1955. Decl. Feb. 14, 1957. 32p. Contract W-7405-eng-92. $0.35(OTS).

Various physical and mechanical properties have been determined for high- and low-C base U, U–Cr, and U–Zr alloys. These data complement those obtained at ANL on the same materials. Their data include studies of radiation damage. On the basis of the properties studied at BMI, the low-carbon Cr and low-carbon Zr alloys appear to be comparable; both are much superior to beta-treated U. The high-carbon Zr alloys are vastly inferior to the low-carbon Zr alloys and also to the high-carbon Cr alloys. High C content does not greatly affect the properties of the Cr alloys. Similarly, high C content in the base U did not appreciably alter its properties, although the direction of growth during thermal cycling was reversed. (auth)

8471 BMI-1002

Battelle Memorial Inst., Columbus, Ohio.

LOW-MELTING ALLOYS FOR CAST FUEL ELEMENTS. Henry A. Saller, Frank A. Rough, and Arthur A. Bauer. May 19, 1955. Decl. Feb. 13, 1957. 18p. Contract W-7405-eng-92. $0.25(OTS).

An investigation has been conducted to determine the composition of uranium-rich ternary eutectic alloys most suitable for reactor application in the as-cast condition. These determinations have been based upon metallographic examination and thermal analysis of as-cast alloys. Of the systems studied, ternary eutectic alloys have been identified in a number of systems, at or near the following compositions: uranium–22.5 at. % chromium–2.5 at. % thorium, uranium–12 at. % silicon–3 at. % aluminum, uranium–14 at. % chromium–4 at. % aluminum, uranium–10 at. % silicon–10 at. % chromium, uranium–10 at. % silicon–2.5 at. % thorium, uranium–11 at. % silicon–4 at. % vanadium, and uranium–9 at. % silicon–6 at. % nickel. The absence of a ternary eutectic in the uranium-rich portion of the following systems has also been established: uranium–silicon–iron, uranium–vanadium–iron, uranium–chromium–vanadium, and uranium–vanadium–thorium. A brief investigation of uranium binary systems has established the uranium–silicon eutectic composition as 8.6 at. % silicon. Data have also been obtained on the uranium–aluminum, uranium–copper, uranium–iron and uranium–tin systems. (auth)

8472 BMI-HRN-3

Battelle Memorial Inst., Columbus, Ohio.

MEMO ON WORK AT BATTELLE DURING DECEMBER 1946. H. R. Nelson. Dec. 31, 1946. Decl. Feb. 14, 1957. 19p. Contract W-7405-Eng-92. $3.30(ph OTS); $2.40(mf OTS).

Status of the work on the extrusion of BeO tubes, thermal conductivity measurements on BeO, compounding concrete for shielding, rolling of Al-clad U–Al strips, protective atmosphere furnace for creep tests of Be, and the construction of a furnace for melting Be are described. (W.L.H.)

8473 BMI-HRN-6

Battelle Memorial Inst., Columbus, Ohio.

MEMO ON WORK AT BATTELLE DURING MARCH, 1947. Mar. 31, 1947. Decl. Feb. 18, 1957. 13p. Contract W-7405-eng-92. $3.30(ph OTS); $2.40(mf OTS).

Progress on extruding BeO cylindrical rods and thick- and thin-walled tubes is summarized. Results of tests carried out to increase the bulk density of iron-aggregate concretes are presented. Samples of 33% U–Al alloy cut from ingots poured at various mold and furnace temperatures were clad and rolled to 0.04-in. sheet. Compound particle formation in low-uranium Al–U alloys was investigated. Creep of Be at 1700°F and 28.7 psi is discussed. Unsoundness in Be castings could not be correlated with the humidity of the protective atmosphere. (L.M.T.)

8474 BMI-T-1

Battelle Memorial Inst., Columbus, Ohio.

A SURVEY OF METALLIC MATERIALS FOR HIGH-TEMPERATURE SERVICE. J. T. Stacy. Apr. 10, 1948. Decl. Feb. 20, 1957. 38p. Contract W-7405-Eng-92. $0.35(OTS).

A brief review is given of materials for high-temperature service which covers particularly the strengths of the various materials at elevated temperatures for service lives of one hundred or one thousand hours. Short discussions of a few high-temperature tests, types of high-temperature alloys, and the dependability and commercial availability of some of these alloys are given. The usefulness of tungsten and molybdenum at high temperatures is considered. Various properties, such as corrosion resistance, weldability, thermal expansion, and deformation, are discussed. Recommendations are made concerning alloys most suitable for service above 1500°F. (auth)

8475 BRB-15

Bridgeport Brass Co., Conn.

THE "SINE PHI" CORRECTION AND ITS APPLICATION TO NORMALIZING ROLLED URANIUM POLE FIGURE DATA. Joseph Singer. Sept. 26, 1955. Decl. Feb. 16, 1957. 14p. Project No. 401.F.3.C. Contract AT(30-1)-1405. $0.25(OTS).

Using the well-characterized 300°-rolled U rod of Chernock and Beck (ANL-4839), their "sine phi" correction is tested by comparing the integrated pole charts for several reflections with the random (Warren's calculated) intensities. This procedure demonstrates the validity of the sine ϕ correction, and is related to material of Williams and Jetter (TID-5151). A simple method for obtaining the defocusing corrections required by Schulz-head data is demonstrated. Also, an alignment gage is mentioned. (auth)

8476 CF-50-5-102
[Oak Ridge National Lab., Tenn.]
FABRICATION OF EXTRUDED URANIUM ALUMINUM RODS FOR CP-3 REACTOR. C. E. Center. May 11, 1950. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The health physics, SF accountability, and security aspects of fuel-element fabrication for the Argonne Heavy Water Reactor (CP-3) are outlined. Slug specifications are included. (C.W.H.)

8477 CF-54-4-224
Oak Ridge National Lab., Tenn.
REPORT OF LITERATURE SURVEY OF BERYLLIUM. J. W. Woods. Apr. 13, 1954. Decl. Feb. 15, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8478 CF-56-3-167
Oak Ridge National Lab., Tenn.
HRP TITANIUM FABRICATION EXPERIENCE FROM LOOP T-1 CONSTRUCTION. P. A. Haas. Mar. 15, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The experience with fabrication of titanium obtained from construction of loop T-1 is summarized for application to other Chemical Technology Division programs. Techniques developed now give acceptable welds without use of an inert gas dry box; this procedure and other improvements will be incorporated in a revised titanium welding procedure to be issued by the ORNL Metallurgy Division. Schedule 80, $\frac{1}{4}$-in.-diam. and $\frac{1}{2}$-in.-diam. titanium pipe was found to have defects; characteristics of the defects and a program for complete inspection of titanium pipe and tubing are being studied. Modification of specifications to permit use of standard grades of commercially pure titanium is suggested for any large future division purchases of titanium. (auth)

8479 CF-56-4-73
Oak Ridge National Lab., Tenn.
TRANSFER OF CARBON BETWEEN DISSIMILAR METALS IN CONTACT WITH MOLTEN SODIUM. TEST CONDITIONS: 1000°C—400 HOURS. R. Carlander and E. E. Hoffman. Apr. 2, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The extent to which AISI-C 1043 steel is decarburized in certain container materials when in contact with molten sodium has been determined at 1000°C for 400 hours. The extent of decarburization of a steel specimen was greater when an Armco iron container was used than when a type 304 elc stainless steel container was used. In addition, sufficient nickel mass transferred to the surface of the AISI-C 1043 steel in the stainless steel test to cause a phase transformation to a depth of 2 to 4 mils. The extent of decarburization of the AISI-C 1043 steel was greater in the 400 hour test than in a previous 100 hour test under similar test conditions. The amount of nickel mass transferred to the surface of the steel specimen was greater in the 400 hour test than in the 100 hour test and furthermore, carburization of the vapor zone of the stainless steel capsule occurred while it did not in the previous 100 hour test. Further tests are to be performed in which the extent of decarburization will be studied when an inert container material (tantalum) and sodium low in oxygen (distilled) will be used. (auth)

8480 CF-56-6-63
Oak Ridge National Lab., Tenn.
MASTER ORDER OF HRP GRADE 347 STAINLESS STEEL. W. R. Gall. June 6, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Proposed ORNL specifications for type 347 wrought austenitic stainless steel are listed. (B.J.H.)

8481 CK-895
California. Univ., Berkeley. Radiation Lab.
MEMORANDUM REPORT ON THE PREPARATION OF FUSED METAL BISCUITS BY HALIDE REDUCTION. Leo Brewer. Aug. 18, 1943. Decl. Feb. 16, 1957. 8p. (A-1166). $1.80(ph OTS); $1.80(mf OTS).

The necessary conditions for the preparation of fused U and Th metal biscuits by reduction of halides are discussed. Pertinent data are presented and the results of calculations of the temperatures attained by various reaction mixtures are given. The possibility of the application of the method to preparation of Pu is considered. Heat content tables for the substances under discussion are included in an appendix. (auth)

8482 CRD-R-2
California Research and Development Co., Livermore, Calif.
THERMAL STRESSES IN CLAD SPHERES. G. W. Brown. Nov. 25, 1952. Decl. Feb. 14, 1957. 12p. (LWS-24651). $3.30(ph OTS); $2.40(mf OTS).

8483 CRD-T2B-49
California Research and Development Co., Livermore, Calif.
TEMPERATURE AND THERMAL STRESS DISTRIBUTION IN A ZIRCONIUM CLAD URANIUM PLATE. N. R. LeRoy and R. L. McKisson. Aug. 15, 1951. Decl. Feb. 14, 1957. 22p. (LWS-23069). $4.80(ph OTS); $2.70(mf OTS).

In the design of a primary target for the deuteron beam, uranium clad with zirconium is being studied as a possible target material. These clad uranium plates are arranged in target tubes such that coolant passes between each plate. Of primary concern are temperature variations and stress distributions resulting from the energy loss in the individual plates. These results are easily obtained for homogeneous plates, and it is desired here to determine the effect of cladding. The derivations of the temperatures and stress distribution equations and an example of the use of these relations follow. (auth)

8484 CT-753
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. June 10, 1943. Decl. Feb. 18, 1957. 29p. Contract [W-7405-eng-92]. (A-931). $4.80(ph OTS); $2.70(mf OTS).

8485 CT-1793
Chicago. Univ. Metallurgical Lab.
TECHNOLOGICAL RESEARCH—METALLURGY QUARTERLY RECORD REPORT ON ELECTROPLATING PERTAINING TO COATINGS FOR TUBALLOY [URANIUM]. A. G. Gray and R. F. Wehrmann. July 11, 1944. Decl. Feb. 18, 1957. 13p. Contract [W-7401-eng-37]. (A-2622). $3.30(ph OTS); $2.40(mf OTS).

Techniques for effecting both adherent and non-adherent electroplates on U are discussed, with emphasis on the pretreatment of base metal. Chromium electroplates appear satisfactory as "blocking layers" to prevent the diffusion of U into the coating during the high-temperature Al—Si hot dip operation. Nickel electroplates on U which were diffusion heat treated showed very good corrosion resistance to pile cooling water. (L.M.T.)

8486 CT-2443
[Chicago. Univ. Metallurgical Lab.]
ELECTROPLATING. Final Report on P. A. No. 154-ML-52-2. Ralph Wehrmann. Dec. 1, 1944. Decl. Feb. 23, 1957. 26p. Contract W-7401-eng-37. $4.80 (ph OTS); $2.70(mf OTS).

A procedure for electroplating U was developed which involves anodic treatment of U in a concentrated solution

of CCl_3COOH prior to plating. Uniform and rapid attack on U, small metal loss, and good plate adherence are advantages of the method. The corrosion resistance of diffused (high temperature) Ni and Ni–Mo plates appears to be 10 to 100 times better than unalloyed U. (auth)

8487 CT-2607
[Chicago. Univ. Metallurgical Lab.]
METALLURGY DIVISION PROGRESS REPORT [FOR] THE PERIOD DECEMBER 30 [1944] TO JANUARY 6, 1945. Jan. 11, 1945. Decl. Feb. 18, 1957. 12p. Contract W-7401-eng-37. $3.30(ph OTS); $2.40(mf OTS).

Progress is reported in the following studies: the molding of U and U oxides; corrosion tests of Nb–U alloys in boiling water; phase studies of Ta–U alloys; the thermal conductivity of Si–U alloys; the microstructure of Al–Mg–U and Rh–U alloys; the preparation of beryllia crucibles; grain structure and hardness of Th samples; the preparation and microstructure of Th–U alloys; the coating of Th with Al by hot-dipping; and the solubility of Ni–Mo alloys in acid solutions. (C.H.)

8488 CT-2951
Iowa State Coll., Ames.
THE CASTING OF THORIUM METAL AND SOME PROPERTIES OF THE CAST METAL. W. H. Keller, Ward Lyon, Harry J. Svec, and Richard Thompson. Oct. 2, 1945. Decl. Feb. 18, 1957. 20p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40(mf OTS).

Methods of casting Th metal produced by the metallothermic reduction of ThF_4 by Ca with various oxidants are described. The metal was cast in vacuo by melting in a BeO crucible at 1800°C, and by permitting it to flow freely into a BeO mold also maintained at 1800°C. Some physical properties of the cast material are presented and the working and shaping of the metal are discussed. (auth)

8489 CT-2958
Iowa State Coll., Ames.
EARLY METHODS FOR CASTING URANIUM AT IOWA STATE COLLEGE. C. F. Gray. July 1945. Decl. Feb. 18, 1957. 45p. Contract W-7405-eng-82. $7.80(ph OTS); $3.30(mf OTS).

Various methods for the casting of uranium and utilizing various graphite molds and crucibles are described. (C.W.H.)

8490 CT-2961
Iowa State Coll., Ames.
THE URANIUM–BISMUTH SYSTEM. D. H. Abmann and R. R. Baldwin. Nov. 12, 1945. Decl. Feb. 18, 1957. 19p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40(mf OTS).

Phase equilibria of the U–Bi alloys have been investigated by thermal and chemical analyses. Two compounds, UBi and UBi_2, were identified; UBi melted above 1500°C and UBi_2 decomposed peritectically at 980°C. There appeared to be an immiscibility gap between 4.9 and 43% Bi. (C.W.H.)

8491 CT-2983
[Battelle Memorial Inst., Columbus, Ohio.]
PROGRESS REPORT ON THE PROPERTIES OF ALLOYS. H. W. Russell, H. R. Nelson, and L. H. Grenell. May 1, 1945. Decl. Feb. 18, 1957. 28p. Contract W-7405-eng-92. $4.80(ph OTS); $2.70(mf OTS).

8492 CT-3054
[Battelle Memorial Inst., Columbus, Ohio.]
THE PROPERTIES OF ALLOYS. Progress Report. H. W. Russell, H. R. Nelson, and L. H. Grenell. June 1, 1945. Decl. Feb. 18, 1957. 28p. Contract W-7405-eng-92. $4.80(ph OTS); $2.70(mf OTS).

The dissociation-pressure curve for ThH_4 in current studies shows some variation from the results previously reported. This may relate to impurities in the sample. Th_3N_4 was formed readily. Studies of the dissociation pressures are inconsistent. Examination of Th–C alloys after heating at 900°C and furnace cooling shows precipitation of carbide. Hardness data on these specimens indicate a solubility limit in the neighborhood of 0.25% C. The scaling of Th in CO is of the same order as in 99.6% A. Corrosion tests in boiling water give some encouragement that an anodizing treatment of Th may reduce the rate of corrosion. Tests at higher temperature are planned. The results of pickling and plating tests are not promising. A bath of fused potassium polyacid fluoride was found to remove scale on Th which was unattacked by other means. The fused salt is quite unstable, however, and does not appear promising for commercial operation. Tests of the solubility of Ce in Al are in agreement with published data. The $CeAl_4$ compound has been found to have a body-centered tetragonal structure. Recoveries of 83% of Ce charged have been obtained upon adding $CeCl_3$ to molten Al using a cryolite slag. The solubility of U in Al appears to be considerably less than 0.16%. The creep of a 17.3% U–Al alloy at 150°C at 3000 psi load attained a steady rate of 0.00004% per hour after 200 hours on test. (auth)

8493 CT-3310
Massachusetts Institute of Tech., Cambridge.
ALLOYS OF URANIUM AND SILICON. I. THE URANIUM–SILICON PHASE DIAGRAM. B. D. Cullity. June 1945. Decl. Feb. 16, 1957. 31p. Contract W-7405-eng-175. $4.80(ph OTS); $2.70(mf OTS).

The uranium-silicon alloy system is composed of four terminal solid solution (alpha, beta and gamma uranium and delta silicon) with very limited solubility ranges and six intermediate phases (epsilon, which contains about 23 at. % Si, U_3Si_2, USi, U_2Si_3, USi_2, and USi_3). Epsilon forms by a peritectoid reaction at 930°C, and the other intermediate phases form directly from the melt or by peritectic reactions at temperatures between 1500°C and 1700°C. (auth)

8494 CT-3511
Chicago. Univ. Metallurgical Lab.
FLEXURE TESTING OF CARTRIDGES. Report on a Phase of Problem Assignment No. 181-ML-55-6 for Period August 8, 1944 to December 23, 1944. George Rosner. June 7, 1946. Decl. Feb. 18, 1957. 29p. Contract W-7401-eng-37. $4.80(ph OTS); $2.70(mf OTS).

The results of flexure tests on large cartridges containing both 2 and 8 in. slugs are reported. The flexibility of the cartridge containing 2 in. slugs approaches that of the empty cartridge. (D.E.B.)

8495 CT-3804
[Massachusetts Inst. of Tech., Cambridge.]
PROGRESS (A-1) REPORT FOR THE PERIOD DECEMBER 1946 TO MARCH 1947. May 28, 1947. Decl. Feb. 16, 1957. 72p. Contract W-7405-eng-175. $12.30(ph OTS); $4.50 (mf OTS).

Solid billets of Th and compacted powdered Th have been extruded at temperatures between 1400 and 1700°F. Copper cans were used, and no evidence of appreciable alloying with the Th was noted. The Ames thorium has better strength and ductility than the extruded powder, but the latter still has good properties. Work on the U–Ta system has been completed. No intermetallic compounds exist, and the solid solubility is very limited. The solubility of Ta in U at 2000°C is about twice as great as W in U. The maximum solid solubility of U in Be appears to be less than 0.03 at.% U. Data on the segregation of UBe_9 from Be in large castings and after extrusion are given. The

solubility of U in a number of low melting metals has been determined up to 900°C. Results so far obtained indicate that Al and Bi give the greatest solubilities. The results of tensile, stress-rupture and creep tests on Be at elevated temperatures have been summarized. (auth)

8496 FMPC-160
National Lead Co. of Ohio, Cincinnati.
SURVEY OF ANALYTICAL, METALLOGRAPHIC, PHYSICAL TESTING, FABRICATION AND MACHINING METHODS FOR THORIUM. E W. Mautz, O. R. Magoteaux, P. Kleinsmith, and F. H. Meyer, Jr. Mar. 20, 1953. Decl. Feb. 18, 1957. 19p. Contract AT-(30-1)-1156. $0.30 (OTS).

A survey of analytical, metallographic, physical testing, fabrication, and machining methods for thorium was conducted at various AEC sites. Data are summarized. (auth)

8497 FMPC-193
National Lead Co. of Ohio, Cincinnati.
LITERATURE SURVEY ON THE CO-REDUCTION PROCESS FOR PRODUCING ZIRCONIUM–URANIUM ALLOYS. W. B. Clymer and O. R. Magoteaux. Apr. 30, 1953. Decl. Feb. 16, 1957. 6p. Contract AT(30-1)-1156. $1.80 (ph OTS); $1.80(mf OTS).

The survey summarizes results found in seven references on the co-reduction of UF_4–ZrF_4 systems to U–Zr alloys. (G.Y.)

8498 FMPC-489
National Lead Co. of Ohio, Cincinnati.
STANDARD OPERATING PROCEDURE FOR FABRICATING N. A. A. ENRICHED SLUGS. (Section 1.4.3.14). Hubert W. Carroll and Charles J. Michel. Oct. 16, 1954. Decl. Feb. 19, 1957. 12p. Contract AT(30-1)-1156. $3.30(ph OTS); $2.40(mf OTS).

Fabrication of enriched U slugs by machining a workpiece to meet specifications from as rolled and heat treated rod stock is reported. (auth)

8499 HW-8879
[Hanford Works, Richland, Wash.]
DIAMETER MEASUREMENTS OF SLUGS BEFORE AND AFTER PROLONGED HEAT TREATMENT. C. A. Bennett and J. J. Lane. Feb. 16, 1948. Decl. Feb. 19, 1957. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Diameter measurements of 25 slugs taken before and after a prolonged heat treatment showed no evidence of differences that would indicate blistering. The average effect of the treatment was different for the various slugs. Large differences in diameter measurements between slugs and significant differences between indices and positions were found. (auth)

8500 HW-20605
[Hanford Works, Richland, Wash.]
CANNING DEPLETED URANIUM. PRODUCTION TEST 313-110-M. Final Report. T. S. Jones. Mar. 27, 1951. Decl. Feb. 19, 1957. 6p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Uranium billets deficient in U^{235} were subjected to the normal successive steps in the slug fabrication process; viz., rolling, machining, and canning. No unusual behavior was observed at any point in the process, thus indicating that the depleted metal from the standpoint of mechanical properties can be considered equivalent to normal uranium. These two uranium billets yielded 47 four-inch slugs. To create a sound basis for comparison in a pile test, two randomly selected normal uranium billets were subjected to parallel operations and 67 four-inch slugs were produced. These slugs were used for comparison pieces to the depleted slugs in the loading pattern. The depleted uranium slugs had a lower reactivity than normal uranium, the deficiency being about 7.4 inhours per 4-inch slugs. (auth)

8501 HW-25928
Hanford Works, Richland, Wash.
IRRADIATED PROCESS SLUG FILM SAMPLING APPARATUS. C. D. Wilson. Oct. 15, 1952. Decl. Feb. 28, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

8502 HW-26426
[Hanford Works, Richland, Wash.]
INITIAL INVESTIGATION OF THE MECHANICAL PROPERTIES OF URANIUM AT ELEVATED TEMPERATURES. R. E. Hueschen and C. H. Pitt. Dec. 2, 1952. Decl. Feb. 28, 1957. 16p. [Contract W-31-109-eng-52]. $0.30(OTS).

The mechanical properties of uranium were determined in tension at temperatures ranging from room temperature to 800°C. The ultimate and yield strength of U in the α phase was found to decrease regularly with increasing temperature. The strength values obtained for U tested in the β phase are somewhat higher than those obtained by extrapolation just below the α–β transformation temperature. There appears to be little variation in the ultimate and yield strength of U within the β region. Per cent reduction in area appears to increase uniformly through the α, β, and γ ranges. The type of fracture changed from a typical brittle fracture at room temperature to a typical ductile fracture at higher temperatures. (auth)

8503 HW-29128
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
ABRASIVE CUTTING OF IRRADIATED URANIUM. Carl L. Boyd. Sept. 17, 1953. Decl. Feb. 19, 1957. 10p. Contract [W-31-109-Eng-52]. $0.25(OTS).

An abrasive cut-off machine development program, based upon a previous experimental model, was undertaken to provide a method for sectioning of irradiated materials without undue personnel exposure or spread of contamination and finally to provide data for the design of an abrasive cutting unit for use in the work cells. A horizontal-feed type cut-off machine, powered by a $7\frac{1}{2}$ hp motor with V-belt drive has been developed for the submerged sectioning of irradiated uranium and is being used successfully to obtain samples for metallurgical, chemical and physical investigations without excessive personnel exposure or spread of radioactive contaminates. (auth)

8504 HW-35457
Hanford Atomic Products Operation, Richland, Wash.
THE OCCLUSION OF HYDROGEN BY URANIUM DURING SEVERAL TYPES OF BETA HEAT TREATMENTS. William E. Ray. Mar. 3, 1955. Decl. Feb. 25, 1957. 5p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

It was demonstrated that carbonate salts add appreciable amounts of H_2 to U during heating. Lead used as a heating medium does not add an appreciable amount of H_2 but permits H_2 removal during the heating. (F.S.)

8505 HW-47015
Hanford Atomic Products Operation, Richland, Wash.
CENTRIFUGAL CASTING FOR PLUTONIUM AND PLUTONIUM-URANIUM PLATES. F. B. Quinlan and J. C. Tverberg. Nov. 28, 1956. Decl. Mar. 13, 1957. 6p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Preliminary investigations involving the centrifugal casting of metal plates suitable for use in a power reactor proved to be successful. Lead plates cast at medium speeds contained very few voids and exhibited a surface which conformed very closely to that of the mold. (auth)

8506 ISC-200
Ames Lab., Ames, Iowa.
INTERIM REPORT OF AN INVESTIGATION OF THE PROPERTIES OF THORIUM AND SOME OF ITS ALLOYS. G. C. Danielson, Glenn Murphy, D. Peterson, and B. A. Rogers. Jan. 18, 1952. Decl. Feb. 27, 1957. 120p. Contract W-7405-eng-82. $0.70(OTS).

An interim report is made on the investigation of the fabrication, physical and mechanical properties of Th and Th alloys. Tabulation is made on the results of corrosion tests and on the metallography of Th alloys. (J.E.D.)

8507 ISC-203
Ames Lab., Ames, Iowa.
SEMI-ANNUAL PROGRESS REPORT IN METALLURGY. For the Period April 1, 1951–September 30, 1951. H. A. Wilhelm. Mar. 31, 1952. Decl. Feb. 27, 1957. 88p. Contract W-7405-eng-82. $13.80(ph OTS); $4.80(mf OTS).

Progress is reported on the treatment of Zr ores; Hf–Zr separation; preparation and reduction of ZrF_4; the preparation, production, and corrosion of Zr alloys; preparation of Th fluoride; reduction of ThF_4; and a study of the metallurgy and mechanical and physical properties of Th. A study is also presented on the properties of V, reduction of the oxide, and properties of Ti–V alloys. (J.E.D.)

8508 ISC-248
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR JANUARY, FEBRUARY AND MARCH 1952. June 20, 1952. Decl. Feb. 27, 1957. 31p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

Progress is briefly reported on the following projects: zircon ore treatment, Hf–Zr separation studies, preparation of Zr by bomb reduction, properties of Zr alloys, corrosion resistance of Zr alloys, Th production, ThF_4 production, Th reductions, melting of Th, effects of impurities in Th, properties of Th alloys, chemistry of Th, recovery of Th turnings, preparation of ZnF_2 and Zn fluozirconate, properties of V alloys, electric conductivity of metals, treatment of Nb and Ta ores, and preparation of rare-earth mixtures. (J.E.D.)

8509 ISC-251
Ames Lab., Ames, Iowa.
QUARTERLY REPORT OF AN INVESTIGATION OF THE PROPERTIES OF THORIUM AND SOME OF ITS ALLOYS. G. C. Danielson, Glenn Murphy, D. Peterson, and B. A. Rogers. July 1, 1952. Decl. Feb. 26, 1957. 50p. Contract W-7405-eng-82. $7.80(ph OTS); $3.30(mf OTS).

The physical and mechanical properties of unalloyed Th are described. The preparation, corrosion, and physical and mechanical properties of the alloys of Th with C, O_2, Ca, Ti, V, Cr, Fe, Zr, Mo, Sn, and Pb are also discussed. (J.E.D.)

8510 ISC-274(Rev.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN ENGINEERING FOR APRIL, MAY AND JUNE 1952. Sept. 20, 1952. Decl. Feb. 27, 1957. 16p. Contract W-7405-eng-82. $3.30(ph OTS); $2.40(mf OTS).

Progress is briefly reported on production of Th from monazite and recovery of spent silica gel from the adsorption separation of HF and Zr. (L.M.T.)

8511 ISC-290(Rev.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR APRIL, MAY AND JUNE 1952. Sept. 20, 1952. Decl. Mar. 4, 1957. 29p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

8512 ISC-531
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR JULY, AUGUST, AND SEPTEMBER 1954. Feb. 17, 1955. Decl. Mar. 28, 1957. 36p. Contract W-7405-eng-82. $0.40(OTS).

The studies on the fusion of niobite-tantalite with NaOH and Na_2O_2, and the separation of Hf from Zr were continued. Initial work was begun on the separation of Mg from Ca metal by sublimation and the extraction of Zr and Mo from molten U with Ag. $ThCl_4$, 99+%, was prepared. Investigations were continued on the preparation of V, Nb, and Ta, and on the phase diagrams of U–Zn, Th–Zr, Hf–Th, C–Th, Nb–U, Ta–Zr, and Zn–Zr. The corrosion of Ta by U–Cr eutectic alloys was described. The measurement of physical properties of Zn single crystals and sodium tungsten bronze crystals by ultrasonic techniques was continued. Intermediate phases in the Al–U system are discussed. X-ray diffractions patterns and resistivity measurements were reported for Ca as part of a study of Ca allotropes. Additional physical constants for Hf metal were presented. Hardness measurements on Zr were correlated while the Zr reclamation study was continued. (C.W.H.)

8513 KAPL-M-JEB-2
Knolls Atomic Power Lab., Schenectady, N. Y.
A MECHANISM FOR THERMAL RATCHETING. J. E. Burke and A. M. Turkalo. Apr. 14, 1951. Decl. Feb. 26, 1957. 21p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

A thermal ratchet theory is proposed and substantiated by experimental evidence. Observations on dimensional changes in U are reviewed and explained, and new predictions are made concerning the behavior of U in thermal cycling. (T.R.H.)

8514 KLX-1721
Vitro Corp. of America, New York.
ELECTROKINETIC PROCESSES—NUCLEAR ASPECTS. QUARTERLY PROGRESS REPORT FOR FEBRUARY 1–APRIL 30, 1954. (Job-1099). May 15, 1954. Decl. Mar. 1, 1957. 23p. Contract AT(30-1)-850, Amendment No. 14. $4.80(ph OTS); $2.70(mf OTS).

The preparation of metallic coatings by electrophoretic deposition of metallic oxides and by subsequent reduction and sintering in purified H was studied. A technique was developed for the electrophoretic deposition of smooth, uniform coatings of SiC, suitable for all metallic carbides. A procedure for the preparation of a ZrH_4-alcohol dispersion with good electrophoretic coating properties was studied, and the application of porcelain enamels by electrophoretic deposition was investigated in an effort to obtain thin uniform ceramic coatings on Al for corrosion protection. Since a monitoring device was necessary for the control of electrophoretic plating, tests were made with two types of automatically recording balances, a strain gage unit and an electromagnetic unit, to weigh continuously the material plated per unit time. A process was developed for the preparation of electrophoretic coating dispersion by a precipitation process. The preparation of UO_2-alcohol dispersion is described. (J.E.D.)

8515 MIT-1004
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
TECHNICAL PROGRESS REPORT FOR THE PERIOD FEBRUARY THROUGH APRIL 1948. A. R. Kaufmann. May 24, 1948. Decl. Feb. 27, 1957. 132p. Contract W-7405-eng-175. $21.30(ph OTS); $6.90(mf OTS).

A method of recovering U from mixtures of BeO and U_3O_8

has been developed. Bottom pouring of Be using a stopper rod inserted from below the crucible is utilized. Radiographic inspection techniques for Be billets are discussed. It is shown that radial plane cracks are revealed by a radial exposure only when the plane of the crack is within an angle of 40° to the x-ray beam. Be flats with a ratio of thickness to width of 3.3 were extruded with reductions of 3, 7, and 10. No appreciable dependence of transverse and longitudinal strength on the amount of reduction was observed. The stress-rupture strength of extruded Be at 1950°F appears to be the same in the transverse and longitudinal directions. Hence, the anisotropy observed in an ordinary tensile test disappears when failure occurs at the grain boundaries in a slow test. The stress-rupture strength at 2000°F of extruded mixtures of Be and beryllia increases continuously with the beryllia content. A detailed study of the Be-Fe phase diagram has been completed. The extrusion of U at temperatures as low as 200°C is shown to be feasible. The metal leaves the machine at a red heat. Microstructures and hardness as a function of extrusion temperature below 600°C are presented. The recrystallization temperature of cold rolled U decreases from about 500°C for a 7% reduction to 450°C at 15% reduction and then appears to remain at this temperature for reductions up to 45%. Mixtures of UBe_9 and of UO_2 with powdered Be may be successfully extruded. Part of the UO_2 is converted to UBe_9 in the process. Crystal bar Zr may be melted in a graphite crucible with only a small C pick-up and with an increase of Rockwell Be hardness from about 20 to 70. Melting in thoria is much less satisfactory since the O and Th pick-up increases the hardness to about 30 on the Rockwell C scale. Zr has been extruded both in the alpha and beta phases with and without an Fe jacket. The extruded Zr has better physical properties than the cast metal. The tensile strength of pure Zr decreases rapidly with temperature, but the ductility increases. A technique for the metallographic examination of Zr is described in detail. Crystal bar Zr appears to increase in weight continuously when heated in air at 500°C, but at 400°C and below the small initial weight gain comes to a constant value after several hundred hours. (D.E.B.)

8516 MIT-1065

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

TECHNICAL PROGRESS REPORT FOR THE PERIOD OCTOBER, 1950 THROUGH MARCH, 1951. A. R. Kaufmann. July 27, 1951. Decl. Feb. 26, 1957. 52p. Contract AT-30-1-981. $7.80(ph OTS); $3.30(mf OTS).

A self-pouring method for making zirconium alloys using the autocrucible is described. A rough determination of the lattice parameters for zirconium–oxygen alloys up to two atomic percent oxygen was made. Various samples of Zr which were high in impurities, such as C, Al, Ti, and N_2, did not corrode appreciably in 350°F water. A preliminary evaluation of the 750°F steam corrosion test indicated that correlation with the 600°F test was not good if the nickel content is high. Preliminary results on the corrosion behavior of alloys of Zr with Ni, Fe, Ti, Al, Ge, V, Cu, and Sn are reported. Crystallite formation and growth of Be during hot working are discussed. Small-scale studies of the extrusion of α uranium showed that a reduction of 25 times at 450°C is possible. The cladding of U bars or tubes with a thin layer of Zr directly by extrusion was carried out successfully. The bond between Zr and U is strong enough to survive bending, swaging, drawing, and thermal cycling operations. (auth)

8517 MIT-1086

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

TECHNICAL PROGRESS REPORT FOR THE PERIOD OCTOBER, 1951 THROUGH MARCH 1952. June 12, 1952. Decl. Feb. 28, 1957. 105p. Contract AT(30-1)-981. $16.80(ph OTS); $5.70(mf OTS).

Single crystals of Be were made by casting the metal into rectangular alundum molds. Polarized light studies of Be single crystals revealed that the optical effect is altered by exposure to N. Be crystals were found to fracture on the basal plane and on second-order prism planes. Ductility in tension was observed for stresses which are oriented in such a way as to avoid fracture. U was extruded at 450°C with a reduction of 25 times. Extrusion cladding of U and U plus 2% Zr with Sn–Zr alloy was developed for rods of about 0.3 in. in diameter. It was determined that α-extruded U sometimes does not grow during thermal cycling. Experiments on heat transfer from U clad with Zr to liquid Pb–Bi are in progress. Successful melting of Ti was carried out in an auto crucible using a graphite retainer. Small additions of Ni and Fe were found to improve the corrosion resistance of Zr to high-temperature steam. Tolerance limits were determined for N in Sn–Zr alloy. The corrosion resistance of Sn–Zr melted in graphite was found to be improved after a fabrication operation. Methods for spectrophotometric determination of Mo and Cr in Zr are reported. The thermal shock resistance of Be was evaluated by determining a number of the pertinent physical constants as a function of temperature. Be and graphite laminates were brazed together with a Zr alloy. A new and simple method of rapidly determining melting points of metals is described. The melting point of crystal bar Th is placed at about 1702°C. Kinetic studies of ordering in Ni_3Mn indicate that both short-range order and anti-phase domains are important. (auth)

8518 MIT-1102

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

TECHNICAL PROGRESS REPORT FOR THE PERIOD APRIL 1952 THROUGH SEPTEMBER 1952. Dec. 10, 1952. Decl. Feb. 26, 1957. 98p. Contract AT(30-1)-981. $15.30(ph OTS); $5.40(mf OTS).

Purification of beryllium by zone melting is being explored. Certain types of high density inclusions in extruded beryllium powder apparently are introduced during the powder compacting operation. A detailed description of the extrusion cladding of small-diameter uranium and uranium-zirconium is given. The results of various heat-treatments of the clad rods on growth in thermal cycling are presented. Thorium has been successfully extruded. The melting of zirconium–tin alloy in graphite crucibles has been studied in a series of 50 melts of 15 to 30 lb size. About 600 ppm of carbon is added to the metal on the average. The corrosion resistance of this alloy in 600°F is satisfactory. Polarized light examination of polished zirconium after corrosion in water gave no obvious correlation with corrosion resistance of nitrogen alloys. (auth)

8519 MIT-1108

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

TECHNICAL PROGRESS REPORT FOR THE PERIOD OCTOBER 1952 THROUGH DECEMBER 1952. Feb. 25, 1953. Decl. Feb. 26, 1957. 161p. Contract AT(30-1)-981. $24.30(ph OTS); $7.50(mf OTS).

Small amounts of Be have been distilled in a vacuum of about 5×10^{-6} mm Hg. Small single crystals of this material were found to give brittle fractures. Some evidence has been obtained indicating that iron in amounts of 2000 to 4000 ppm improves the corrosion resistance of Be in 250°C water. The present technique for extruding Be powder is described in detail and the effect of extrusion variables on ductility and hardness is described. Polarized light may be used to

obtain information on preferred orientation in Be. Pechiney Be has been found to be very pure, but the ductility of extruded rod was not high. The operation of an internally cooled iron pipe in molten U–Fe eutectic alloy is described. Preliminary attempts to cast U–Fe alloy into iron-tube radiators have given encouraging results. Several extrusions of U and Zr have been carried out to give multiple cladding of subdivided U. It appears that homogeneous U–Zr alloys can be made in graphite crucibles, without going to high temperatures, by the use of a low-melting Zr–Zn alloy. Extensive tests continue to show that graphite-melted Zr alloys, using as little as 1% Sn or Ni, are corrosive-resistant to 600°F water. A preliminary metallurgical study of Al alloys, with small amounts of Mg or Zr, has been carried out. Zr is particularly effective in strengthening the Al. Shell molding is being successfully used for casting extrusion dies of refractory alloys. (auth)

8520 MIT-1111

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

TECHNICAL PROGRESS REPORT FOR THE PERIOD JANUARY 1953 THROUGH MARCH 1953. June 12, 1953. Decl. Feb. 28, 1957. 91p. Contract AT(30-1)-981. $13.80(ph OTS); $4.80(mf OTS).

Be powders were hot-compacted to form 3½-in. billets at temperatures between 400 and 1000°C and then recanned and extruded. Results indicate that high-ductility metal can be made in this way but the procedure requires further study. Some physical properties of the U–Cr and U–Fe eutectic alloys have been determined. These include thermal expansion up to the melting point, thermal conductivity, and density. Casting of the U–Cr alloy around closely spaced iron tubes has been successfully carried out using a thin oxide coating on the iron. It appears that iron may be removed from a U–Cr–Fe alloy by oxidizing the alloy with steam and using magnetic separation. A careful evaluation of the physical properties of graphite-melted Zr–Sn alloys has been completed. It appears that oxygen decreases the ductility of such alloys less than for Zr alone. Various heat treatments seem to have little influence on the corrosion resistance of the material but it is found that quenching from the beta phase increases the yield and ultimate strengths. It has been found that hydrogen embrittles Zr–Sn alloys and that a notched tensile test will reveal the effect. Graphite melting in vacuum reduces the hydrogen content below the embrittlement point. Metallographic results on Zr–Hf alloys with some observations on the recrystallization and transformation temperatures are reported. Highly successful procedures for grinding of Zr, Be, and U alloys are described. (auth)

8521 MIT-EJF-2

Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.

PROGRESS REPORT OF BROOKHAVEN WORK BEING CARRIED OUT AT MIT. E. J. Field, Jr. Sept. 14, 1948. Decl. Feb. 26, 1957. 16p. $3.30(ph OTS); $2.40 (mf OTS).

In a study of anodizing Al–U slugs, the Al–U reaction was found to be much greater in unanodized slugs. No detrimental effect of anodizing to creep was found, and higher temperature did not increase the reaction. (F.S.)

8522 N-1779

Chicago. Univ. Metallurgical Lab.

NOTES TAKEN AT THE METALLURGY CONFERENCE ON DECEMBER 14–15, 1944. Paul Lauletta. Dec. 21, 1944. Decl. Feb. 26, 1957. 14p. (MUC-ABG-310). $3.30(ph OTS); $2.40(mf OTS).

Metallurgical studies on U and Th and their alloys being conducted at Ames, Battelle, Chicago, M.I.T., and Westinghouse are reviewed. (D.E.B.)

8523 NMI-1128

Nuclear Metals, Inc., Cambridge, Mass.

EVALUATION OF UNIFORMITY OF URANIUM CONCENTRATION IN TWO COEXTRUDED ALUMINUM–URANIUM ALLOY TUBES BY SPECIFIC VOLUME AND CHEMICAL ANALYSIS METHODS. J. L. Klein and L. R. Aronin. Jan. 20, 1955. Decl. Feb. 28, 1957. 26p. Contract AT(30-1)-1565. $0.35(OTS).

Analytical expressions are given for calculating uranium concentration in gm U/cm^2 and equivalent core composition for two methods of evaluation: from mass density measurements of clad samples, and from colorimetric chemical analysis of clad samples. Analysis of errors inherent in each method is discussed. Ten samples from each of two tubes were studied for uniformity of U content. The tubes were chosen without regard to absolute uranium content of the core castings. The variation in gm U/cm^2 was within the tolerance of ±5% of the average value for each tube. Good agreement was obtained between the density and chemical analysis methods of evaluation. The tube representing a core casting having a good radiography rating showed less variation in calculated core composition than the tube representing a core casting having a poor radiograph. Billet radiography, therefore, gives meaningful comparisons. (auth)

8524 NMI-2036

Nuclear Metals, Inc., Cambridge, Mass.

FUNDAMENTAL AND APPLIED RESEARCH AND DEVELOPMENT IN METALLURGY. Progress Report for July 1955. July 29, 1955. Decl. Feb. 27, 1957. 10p. Contract [AT(30-1)-1565]. $0.25(OTS).

A brief review of the progress of the work on extrusion of U–Zr alloys, Th (Al clad), and zirconium (Cu clad) is presented. A study was made of the effect of cooling on the corrosion of Sn–Zr alloys and mass transfer of V in liquid Bi. (For preceding period see NMI-2036.) (J.E.D.)

8525 NYO-1119

Brush Beryllium Co., Cleveland.

QUARTERLY PROCESS DEVELOPMENT PROGRESS REPORT FOR JANUARY, FEBRUARY, AND MARCH 1953. A. J. Stonehouse, comp. May 15, 1953. Decl. Feb. 28, 1957. 31p. Contract AT(30-1)-541. $4.80(ph OTS); $2.70(mf OTS).

Current work on the development of improved techniques for the fabrication of Be by powder metallurgy including the manufacture of Be tubing and the direct pressing of Be-powder to shape is described. Summaries are also provided for the current work on the development of a shock-resistant Be rod, Be process metallurgy, and the effect of vibrational energy upon U metal when applied during the solidification process. (auth)

8526 NYO-1327

Mallinckrodt Chemical Works, St. Louis.

PRELIMINARY REPORT ON VACUUM CASTING OF URANIUM INTO HEATED MOLDS. W. H. Keller. Dec. 17, 1951. Decl. Mar. 2, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

Cast U with a carbon content as low as 220 ppm, nitrogen content as low as 20 ppm, and density as high as 19.00 g/cm^3 at both the bottom of the ingot and 5 in. from the top and with excellent surface and free of pipe below 1 in. from the top has been produced by casting U into heated graphite molds. Melts have been poured at 2400°F (thermocouple temperature in the base of the crucible) and the surfaces

obtained indicate that pouring temperature can be dropped even lower (auth)

8527 NYO-1328
Mallinckrodt Chemical Works, St. Louis.
SUPPLEMENT TO PRELIMINARY REPORT ON VACUUM CASTING OF URANIUM INTO HEATED MOLDS. Supplement to NYO-1327. W. H. Keller. Dec. 26, 1951. Decl. Mar. 2, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

Photographs are presented of ingots produced in the casting of U into heated molds. The photographs are discussed in NYO-1327. (J.E.D.)

8528 NYO-1346
Mallinckrodt Chemical Works, St. Louis.
ATTEMPTS TO MANUFACTURE URANIUM – ZIRCONIUM ALLOYS BY CO-REDUCTION OF THEIR TETRAFLUORIDES. L. G. Weber. Sept. 23, 1952. Decl. Feb. 28, 1957. 9p. Contract W-14-108-eng-8. $0.25(OTS).

A series of tests were made to determine whether alloys of U and Zr could be produced by the co-reduction of U and Zr tetrafluorides with Mg in normal U production bombs using Mg fluoride as a liner. No operational difficulties were encountered but the metal produced showed stratification in the low ranges of Zr content (0.2 to 0.4%) and was heterogeneous in the higher ranges (up to 5%) of Zr content. Operational and analytical data are included. (auth)

8529 NYO-5117
Mallinckrodt Chemical Works, St. Louis.
IMPROVED INTERNAL ASSEMBLY FOR A RECASTING FURNACE. William A. Oppold. May 15, 1945. Decl. Mar. 7, 1957. 8p. Contract W-7405-eng-29. $1.80(ph OTS); $1.80(mf OTS).

Improvements were made on the internal construction of a recasting furnace, in order to increase the vacuum and thereby raise the casting yield, and to improve the control over the heating cycle of the furnace. (J.E.D.)

8530 ORNL-766
Oak Ridge National Lab., Tenn.
EXTRUSION OF 2% U–Al ALLOY FUEL RODS FOR THE ARGONNE CP-3 REACTOR. W. D. Manly. July 26, 1950. Decl. Mar. 8, 1957. 20p. Contract W-7405-eng-26. $0.30(OTS).

The preparation of Al–U alloy fuel rods for the Argonne Heavy Water Reactor is described. Alloy billets were prepared by melting Al and adding metallic U to the superheated Al bath prior to casting in graphite molds; the billets were extruded and then rolled to specifications. Photomicrographs are included. (C.W.H.)

8531 ORNL-1991
Oak Ridge National Lab., Tenn.
HANDLING TECHNIQUES FOR RUBIDIUM. H. E. McCoy, Jr. and E. E. Hoffman. Dec. 12, 1955. Decl. Mar. 1, 1957. 15p. Contract W-7405-Eng-26. $0.30(OTS).

Procedures are outlined for opening Rb containers received from suppliers, distillation and filtration of Rb, and loading liquid Rb in corrosion-testing systems. The corrosive effects of liquid Rb on Inconel at elevated temperatures (~1500°F) were investigated. Physical properties of Rb are tabulated. (C.W.H.)

8532 SEP-18
Sylvania Electric Products Inc., Bayside, N. Y.
THE PRODUCTION OF URANIUM BARS BY POWDER METALLURGY. Norman P. Pinto and Henry H. Hausner. Nov. 3, 1949. Decl. Mar. 5, 1957. 15p. $3.30(ph OTS); $2.40(mf OTS).

Uranium was compacted at pressures from 25 to 100 tsi and sintered at temperatures from 1050 to 1135°C. Bars may be produced exhibiting 95% and over of theoretical density and physical properties, e.g. hardness and electrical resistivity, comparable to those of reguline uranium. Techniques are described for pressing, sintering, and handling of compacts. The metallographic methods developed for examining compacts of uranium are included. (auth)

8533 TID-5246
Horizons, Inc., Cleveland.
INVESTIGATIONS FOR THE PRODUCTION OF THORIUM METAL. Final Report [for] May 1, 1952 to April 30, 1954. B. C. Raynes, J. C. Bleiweiss, M. E. Sibert, and M. A. Steinberg. Nov. 1, 1954. Decl. Feb. 28, 1957. 165p. Contract AT(30-1)-1335. $0.85(OTS).

A research program carried out to develop a new electrolytic process for the preparation of pure thorium metal is reported. During the course of this investigation, two practical methods were developed for the preparation of thorium by fused-salt electrolytic means. One was a modification of a previously reported method, and the other a new development. The former method was the electrolysis of $KThF_5$ in sodium chloride. This double fluoride had previously been used in mixtures of sodium chloride and potassium chloride. The simpler electrolyte behaved as well as, or better than, the eutectic mixture previously used. However, this process has the disadvantage of limited life of the electrolyte. The refractory fluorides accumulate in the melt, since only chlorine gas and thorium metal are produced in the process. These refractory fluorides serve to raise the melting point of the bath and to decrease the electrical conductivity of the melt to a point where further electrolysis is impractical after three to four cathode withdrawals. The new process for the electrolytic production of thorium metal is based upon an all-chloride electrolytic system. In this system $ThCl_4$ dissolved in sodium chloride is employed in a process which produces thorium metal at the cathode, chlorine gas at the anode, and has, therefore, the ability to operate indefinitely. For both processes, electrolytic cells of special construction were used to provide both an inert cell atmosphere and also corrosion resistance to the reactants and to the product. Such cells employ all graphite internal construction wherever possible to eliminate any corrosion problem. Thorium chloride can be prepared by wet chemical procedures from thorium nitrate or from thorium oxycarbonate by treating an HCl solution of these salts with ammonium chloride to produce ammonium thorium chloride. This is followed first, by evaporation and drying, and then by sublimation of the ammonium chloride in the presence of sodium chloride to produce the all-chloride source electrolyte. The purity and physical properties of the product thorium metal are largely dependent upon the purity of the thorium chloride, particularly with respect to oxide and moisture content. High-purity thorium metal was repetitively produced from both source electrolytes. The thorium could be readily arc melted, rolled, drawn, and swaged. Thorium metal with an as-cast hardness of Rockwell B 5 (Vickers 78) has been produced. Complete data on both processes are given. Included also is the design of the laboratory cell and an intermediate size cell used in an expansion of the all-chloride process to a larger scale. Operational data, preparation of source electrolytes, physical data on the melts, and procedures for recovery and evaluation of the thorium metal produced are also included. This report also includes a literature summary of thorium metal preparation and a summary of the properties of some compounds of the metal. (auth)

8534 WAPD-125
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
STRUCTURAL CHANGES IN URANIUM-12 w/o MOLYBDENUM ALLOY DURING EXPERIMENTAL FUEL ROD

FABRICATION. J. G. Goodwin. Jan. 26, 1956. Decl. Feb. 7, 1957. 29p. Contract AT-11-1-GEN-14. $0.35 (OTS).

A proposed method for the production of experimental Zircaloy-2 clad fuel rods with U–12 wt.% Mo as the fuel involves a number of melting and fabrication steps which change the grain structure of the fuel alloy. These steps include all or some of the following: induction melting, consumable electrode arc melting, fuel ingot extrusion, composite billet extrusion (for cladding), cold or hot drawing, cold or hot swaging, annealing, and straightening. This report presents some of the structural changes encountered in the U–12 wt.% Mo fuel alloy from the induction cast ingot through the various stages of fabrication. (auth)

8535 WAPD-135

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

EXTRUSION TECHNIQUES AND BILLET DESIGN FOR THE PRODUCTION OF ZIRCALOY-2 CLAD URANIUM-12 w/o MOLYBDENUM FUEL RODS. J. G. Goodwin, F. R. Lorenz, E. L. Richards, and R. W. Tombaugh. Jan. 10, 1956. Decl. Feb. 7, 1957. 42p. Contract AT-11-1-GEN-14. $0.40(OTS).

A résumé of the work performed toward the development of extrusion techniques and billet design for the production of Zircaloy-2 clad U–12 wt.% Mo fuel rods. Fabrication techniques for cladding components and the fuel alloy are presented. Billet design, extrusion equipment, and factors affecting the extrusion constant are discussed. (auth)

8536 WAPD-FE-407

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

HOT STRAIGHTENING OF REFERENCE FUEL ELEMENT ROD BY VERTICAL AND HORIZONTAL STRETCHING. H. J. Synder. Sept. 13, 1954. Decl. Mar. 6, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

Straightness, sufficient for subsequent finishing operations, was induced in short lengths of severely curved co-extruded Zircaloy-2-clad U–Mo fuel rods by resistance heating them to 1650°F and simultaneously subjecting them to stretching either on a hydraulic draw bench or by vertically suspending with weight attached. (auth)

8537 WAPD-FE-739

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].

PRELIMINARY EXTRUSION PROCESS AND TOOLING SPECIFICATIONS. F. R. Lorenz. Apr. 5, 1955. Decl. Mar. 6, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

Preliminary process specifications for co-extrusion of Zircaloy-2 clad 12 wt. % Mo–U rod are presented. (auth)

8538 WAPD-MDM-10

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

EFFECTS OF NEUTRON BOMBARDMENT UPON THE PROPERTIES OF ZIRCALOY-2. M. L. Bleiberg. May 19, 1954. Decl. Mar. 5, 1957. 17p. Contract AT-11-1-gen-14. $3.30(ph OTS); $2.40(mf OTS).

Changes in properties of Zircaloy-2 were determined after irradiations at process water temperatures in the MTR. The electrical resistivity of this material (measured at −196°C) exhibited an increase which was 15% higher than its pre-irradiation value for the entire range of exposures. The Zircaloy-2 irradiation-hardened to DPH 231, an increase of 33 DPH for the same range of exposures. The notch toughness of irradiated Zircaloy-2 appeared to decrease slightly at lower testing temperatures and increase slightly at higher testing temperatures; also, the notch toughness was not a function of integrated fast flux over the range of exposures investigated. (auth)

8539 WAPD-T-283

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

FABRICATION OF EXPERIMENTAL U–Mo FUEL RODS BY THE DRAWING OF CO-EXTRUDED ROUND RODS. H. J. Snyder. Oct. 13, 1955. Decl. Feb. 7, 1957. 19p. Contract AT-11-1-GEN-14. $0.30(OTS).

Fabrication of 0.300-in. diameter fuel rod stock having U–12 wt.% Mo fuel and 0.030-in. thick Zircaloy-2 cladding by the drawing of co-extruded round rods is discussed. The data presented indicate that both cold (room temperature) and hot (1900°F and 1300°F) drawing can be utilized to produce 0.300-in. diameter fuel rod from 0.437-in. to 0.625-in. diameter extruded rounds. This represents cross sectional reductions of 53% to 77% respectively. Experimental material yields range between 82% and 92%, but indications are that increased material yields would result if proper tooling and longer co-extruded rounds were employed. Molybdenum disulfide in lacquer is a satisfactory lubricant for cold drawing, and a 50-50 wt.% mixture of graphite and table salt suspended in SAE-10 motor oil is a suitable lubricant for hot drawing. Individual drawing reductions of approximately 30% are necessary in order to consistently perform drawing and repeatedly produce a usable configuration between the fuel and the cladding. (auth)

8540 WAPD-TN-523

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

AN X-RAY STUDY OF Zr–U_3Si DIFFUSION ZONES. R. B. Roof, Jr. Nov. 18, 1955. Decl. Mar. 8, 1957. 12p. Contract AT-11-1-GEN-14. $1.80(ph OTS); $1.80(mf OTS).

The migration of Si out of the core of U_3Si to form a metallic silicide with the consequent development of a highly corrodible region of alpha U suggests that the corrosion resistance of U_3Si can be materially improved by the inclusion of a barrier between the U_3Si core and the Zircaloy-2 cladding. The primary requirement of such a barrier is that it does not form a silicide. In this manner the silicon atoms would be contained in the core and the chances for the formation of alpha U would be greatly reduced. The corrosion data on extruded Zircaloy-2 clad U_3Si with barrier layers of Nb and Mo show that all samples fail in 0 to 14 days in 650°F water and that the method of failure is the same as the direct bonded Zircaloy-2 clad U_3Si, i.e., rapid attack at the core-clad interface. Available data on Zircaloy-2 clad U_3Si with a barrier layer of Al indicate that no failures occur due to core-barrier attack and that such failure should not occur as Al does not form a silicide. This is in agreement with the theory postulated and suggests that the most promising avenue of approach in minimizing bondline attack in high temperature water is to employ a barrier layer, such as aluminum, between U_3Si and Zircaloy-2 cladding. (auth)

PHYSICS

8541 ANL-4552

Argonne National Lab., Lemont, Ill.

PHYSICS DIVISION REPORT FOR OCTOBER AND NOVEMBER 1950. Dec. 8, 1950. Decl. Feb. 13, 1957. 57p. Contract W-31-109-eng-38. $9.30(ph OTS); $3.60(mf OTS).

Neutron resonances and cross sections are listed for various elements. Neutron flux measurements were made in the Argonne Heavy Water Reactor using activated Au foils. Information on the crystal structure of Pa, F, and

Pu compounds is reported. Equations are given for the determination of heat flux in a fuel rod from the measured neutron flux. Evidence is given for the dependence of the shapes of the neutron capture γ ray spectra on the odd-even character of the compound nuclei formed after neutron capture. A study was made on radioactive recoil, resulting from radiative neutron capture, from In metal surfaces. A review is made of the literature concerning promising models of the cooler giant stars, and the ranges of temperature and pressure are given for one model. The range of temperatures and densities in the envelopes of several giant stars were found, and absorptive opacity was computed. It is concluded that for some giants and all cepheids a model which assumes the opacity to be due only to scattering is not valid. The effect of a non-absorbing reflector on a resonance absorbing medium is discussed in terms of eigenfunctions relevant to the problem. Calculations are also made of electron scattering in an arbitrary magnetic field. (B.J.H.)

8542 ANP-71(Del.)
Oak Ridge National Lab., Y-12 Area, Tenn.
THERMAL CONDUCTIVITY OF STEEL WOOL AND SOME GRANULAR SOLIDS. D. F. Salmon and J. F. Bailey. Oct. 12, 1951. Decl. with deletions Feb. 19, 1957. 15p. Contract W-7405-Eng-26. $3.30(ph OTS); $2.40(mf OTS).

This report presents results obtained while investigating thermal conductivity of steel wool, granular aluminum oxide, granular silicon carbide and a mixture of granulated silicon carbide and graphite powder considered for use in heat exchangers. Description of test apparatus, method of calculation, and evaluation of results are included. (D.E.B.)

8543 CF-2926
[Chicago. Univ. Metallurgical Lab.]
PHYSICS DIVISION REPORT FOR MONTH ENDING APRIL 15, 1945. A. J. Dempster. May 9, 1945. Decl. Mar. 16, 1957. 36p. Contract W-7401-eng-37. $6.30(ph OTS); $3.00(mf OTS).

Procedures for preparing diborane and removing ethane from diborane are given. Geiger-Mueller tubes operating at very low potentials are described. Lattice dimensions and calculated densities are given for Np_2S_3, U_2S_3, Th_2S_3, Nb_2O_5, and Ta_2O_5. Lattice dimensions are also given for heavily irradiated and control samples of graphite. Some discussion on breeder reactors is given. A resonance formula for nuclear reactions was derived, assuming that the wave function of the compound nucleus independent of energy. The effect of bubble formation on k loss in reactors was studied. Critical size formulas and fertility and heat transfer calculations relative to resonance breeders are discussed. Neutron diffusion in finite bodies is considered by the statistical distribution of chord lengths. The mass transfer coefficient for water through a sphere bed of benzoic acid was studied. Formation of bubbles in supersaturated water during flow through pipes was observed. (L.M.T.)

8544 LWS-22659
California Research and Development Co., Livermore, Calif.
INVESTIGATION OF NO. 1 INCONEL DRIFT TUBE CRACKS. W. B. Myers and J. C. Ekvall. June 5, 1953. Decl. Feb. 27, 1957. 28p. (P-A-6). $4.80(ph OTS); $2.70(mf OTS).

At the completion of an accelerator run one of the Inconel drift tubes was cracked in three places. The operation records were studied to determine the conditions that caused the failure. It was found that the failure was probably due to thermal or residual stresses caused by the beam or sparking. A metallurgical analysis and a thermal stress investigation showed that the cracks were most likely caused by beam heating. The thermal stresses due to sparking occur locally on the surface, and it is doubtful that they could crack the Inconel throughout the thickness A comparison between Inconel and copper showed that materials that have a high thermal conductivity and are very ductile would be less likely to crack due to beam heating. Care must be taken in future Inconel fabrication because the improper fabrication procedures can weaken the material. (auth)

8545 LWS-22702
California Research and Development Co., Livermore, Calif.
EXPERIMENTAL DETERMINATION OF SOME CHARACTERISTICS OF PACKED SPHERE BEDS. J. W. Shortall. Aug. 19, 1953. Decl. Feb. 27, 1957. 25p. (D-T-21). $4.80(ph OTS); $2.70(mf OTS).

Experiments were performed to determine some fundamental characteristics of packed sphere beds to apply to MTA lattice and target concepts as well as to possible reactor designs. Data are given on over-all porosities which occur in cylindrical containers, over-all porosities in a model of a rectangular lattice section and variations in porosity within a rectangular lattice section. A short discussion is given of the bridging effect, or clogging, of spheres when attempt is made to make them flow through cylindrical containers. Included are some experimental data. Data and curves are also given for pressure drops obtained by pumping high velocity water through packed sphere beds in rectangular and cylindrical containers. A comparison of the present investigation with previously published data is given. A bibliography, representative of work done in these fields is included. (auth)

8546 LWS-29063
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
ACCOUNTABILITY, SECURITY, AND HEALTH PHYSICS PROCEDURES FOR Pu^{239} IN EXPONENTIAL STUDIES. Apr. 28, 1953. Decl. Feb. 27, 1957. 18p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

Equipment for an exponential study of the effects of structural materials and lattice design on neutron economy was assembled as part of a design study on fast breeder reactors. The Pu assembly was driven by a water boiler neutron source. The quantities of Pu^{239} needed, and accountability, security, and health physics procedures for Pu^{239} control during the studies are discussed. (C.H.)

8547 NAA-SR-Memo-70
North American Aviation, Inc., Downey, Calif.
PROGRESS REPORT FOR JUNE 1951. Aug. 15, 1951. Decl. Feb. 28, 1957. 22p. Contract AT-11-1-gen-8. $4.80(ph OTS); $2.70(mf OTS).

Work is reported on the Laplacian calculations for the NPR, reactor performance of the Optimized Pu Power Reactor, analysis of breeding in an outside blanket in the High Temperature Power Plant, radiation damage in graphite impregnated with U, and target design for MTA. (F.S.)

8548 ORNL-576
Oak Ridge National Lab., Tenn.
PHYSICS DIVISION QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 15, 1949. Feb. 7, 1950. Decl. Mar. 8, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Radiation of Cu–Zn and Cu–Al Alloys. Pile radiation was found to cause a slight decrease in the resistivity of some of these alloys; this change does not anneal out below 200°C. Bonded-wire Strain Gages. A report has been issued (ORNL-518) covering a study of the best way to use bonded-

wire strain gages for measurement of thermal coefficients of expansion. Applied to uranium, the gages indicate coefficients which vary from specimen to specimen, and even from place to place on single specimens, within the limits of 5 to 30 × 10^{-6}. Improvements on the technique of using these gages continue to be made. X-ray Diffraction. The structures of the U–Al compounds are being investigated. Results on "UAl_3" have revealed the positions of the uranium atoms. Density considerations suggest that "UAl_5" is really UAl_4. Creep Experiments. Fission heaters have been fabricated and tried in the pile, but do not look very promising. Construction proceeds on other apparatus for creep tests on wires and rods. Thermal Conductivity of Uranium-bearing Materials. Preliminary results show a decrease by a factor of 75 in the thermal conductivity of the outer part of a U-graphite slug as the result of pile exposure Radiation Damage in Germanium. Rough agreement has been obtained between the rate of change of conductivity observed in a Ge plate exposed to pile radiation, and the rate to be predicted on a simplified theory. The number of displacements resulting from elastic collisions and the resulting ionization by germanium knock-ons is estimated on stopping-power theory. (auth)

8549 UCRL-2200

California. Univ., Berkeley. Radiation Lab.

MONTHLY PROGRESS REPORT NO. 120 [FOR], MARCH 15, 1953 TO APRIL 15, 1953. May 1, 1953. Decl. Mar. 6, 1957. 18p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Progress in experimental and theoretical physics, MTA target development, accelerator construction and operation, chemistry, biology and medicine, and plant and equipment studies is reviewed. (D.E.B.)

ELECTRICAL DISCHARGE

8550 BNL-2222

Brookhaven National Lab., Upton, N. Y.

THE MAGNETO-HYDRODYNAMICS EQUATIONS OF MOTION OF A COMPLETELY IONIZED PLASMA. Hartland S. Snyder. Feb. 24, 1955. Decl. Feb. 14, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

The magneto-hydrodynamic equations of motion of a completely ionized plasma are developed. The plasma is treated as a mixture of a positively charged fluid and a negatively charged fluid. Viscous forces between the fluids are omitted. (F.S.)

INSTRUMENTS

8551 A-3219

Columbia Univ., New York. Div. of War Research.

THERMAL FLOWMETERS. C. D. Swartz. Jan. 15, 1945. Decl. Feb. 12, 1957. 73p. Contract W-7405-eng-50. (1C-M470). $12.30(ph OTS); $4.50(mf OTS).

The design, calibration, and operating characteristics of a thermal flowmeter for the measurement of gas flow are given. (D.E.B.)

8552 K-307

Carbide and Carbon Chemicals Corp. [K-25 Plant], Oak Ridge, Tenn.

FLUOROTHENE DIAPHRAGMS FOR ONE-HALF INCH HILLS-McCANNA VALVES. A. D. Ferrer and J. Zasler. Nov. 19, 1948. Decl. Feb. 25, 1957. 14p. Contract [W-7405-eng-26]. $0.30(OTS).

The design, construction, and testing of a fluorothene diaphragm to operate in a ½-in. Model 500 Hills-McCanna Valve with a minimum life of 3500 cycles is discussed. A 1/64-in. fluorothene diaphragm softened in trichloroethylene and backed up by the rubber diaphragm supplied with the valve qualified as satisfactory on a basis of flow resistance and endurance in both stainless steel and glass-lined valve bodies. (D.E.B.)

8553 Y-515

Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.

PRECISION SELECTOR FOR VOLTAGE-REGULATOR TUBES. O. B. Rudolph. Nov. 4, 1949. Decl. Mar. 6, 1957. 12p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Investigation of problems of the maintenance of a steady well-regulated accelerating voltage in the operation of a calutron has indicated that 90% of the troubles were caused by faulty voltage-regulator tubes. Development of a precision selector to sweep tubes through the entire operating current range while employing a sensitive detecting device to indicate voltage variations is described. (auth)

ISOTOPES

8554 CF-56-4-185

Oak Ridge National Lab., Tenn.

ESTIMATE OF THE COST OF PRODUCING NITROGEN 15 BY CHEMICAL EXCHANGE: DETERMINATION OF MINIMUM COST NITROX SYSTEM. B. B. Klima. Apr. 25, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An estimate was made of the cost of N^{15} based on the nitrox system. It is the purpose of this cost estimate to determine the system conditions which will give a minimum cost of producing N^{15}. The separation factor (alpha) and the concentration of product H_2SO_4 both vary as the molarity of the HNO_3 is varied. Estimates were made of three systems using 7, 10, and 15 molar HNO_3 as the aqueous phase. One gram of N^{15} product produced at the rate of one pound N^{15} per day using 7 molar HNO_3 will cost $3.93 which is 10% less than the cost using 10 molar HNO_3 and much less than the costs using 15 molar acid. The plant costs are considerably lower for the 7 molar process than for the other two as noted. (auth)

ISOTOPE SEPARATION

8555 A-530

[Columbia Univ., New York. Div. of War Research.]

THE DESIGN OF THERMAL DIFFUSION CASCADES FROM EXPERIMENTAL DATA. Karl Cohen. Feb. 4, 1943. Decl. Feb. 12, 1957. 16p. OEMsr-412. (4-R-20). $3.30(ph OTS); $2.40(mf OTS).

8556 A-2357

[Standard Oil Co. of Indiana, Chicago.]

AN INTERPRETATION OF THE ISOTOPIC SEPARATION IN THE DISTILLATION OF METHYL ETHER–BORON FLUORIDE COMPLEX. D. A. McCaulay. Oct. 3, 1945. Decl. Mar. 12, 1957. 28p. Contract W-7418-eng-41. $4.80(ph OTS); $2.70(mf OTS).

In order to calculate an isotopic separation factor for the equation,

$$B^{10}F_3(g) + Me_2O \cdot B^{11}F_3(l) \rightleftharpoons B^{11}F_3(g) + Me_2O \cdot B^{10}F_3(l),$$

which represents the equilibrium occurring in one stage of a new rapid method for B^{10} analysis, it was necessary to consider the mechanism by which isotopic fractionation occurred in the distillation of the methyl ether–boron fluoride complex. In view of the fact that the B^{11} isotope has been shown to be more volatile than the B^{10} isotope in

all distillations of B compounds in which isotopic separation was observed, it was postulated that separation is due to a chemical exchange reaction and that the difference in vapor pressures between the isotopic compounds is negligible. Expressions were set up for the equilibrium constant K of the above reaction which, when combined with similar expressions for the one plate fractionation factor α, led to the expression,

$$K = 1 + \frac{\alpha - 1}{\text{Fraction Dissociated}},$$

from which K is calculated to be 1.027 when $\alpha = 1.016$. These assumptions were substantiated by laboratory experiments which showed an extremely small difference in boiling points between B^{10} complex and B^{11} complex, and which indicated a value of $K = 1.025 \pm .002$. The value $K = 1.027$ was, therefore, accepted as the desired factor for correcting rapid B^{10} analyses. (auth)

8557 ORNL-1874
Oak Ridge National Lab., Tenn.
CHEMICAL SEPARATION OF ISOTOPES SECTION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1954. May 20, 1955. Decl. Mar. 2, 1957. 32p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

New systems involving the exchange of B between boron trifluoride and boron trifluoride addition compounds have been explored. These systems have large separation factors and potentially simple reflux mechanisms. A precise determination of the separation factor for the anisole–boron trifluoride system gave the value $\alpha_{(30°C.L.)} = 1.0291 \pm 0.0009$. B exchange was found to occur between BF_3 and BCl_3. Several homogeneous catalysts have been found which activate the hydrogen–water exchange, but none are adaptable to the production of deuterium because of the slow exchange rate. Pt of platinum oxide may be usable as a heterogeneous catalyst with proper support or dispersion techniques. The high-pressure solubility of H in several amalgams was investigated in connection with a unique countercurrent exchange system. A proposed system involving isotopic exchange between lithium dipivaloylmethane in diethyl ether and lithium hydroxide in aqueous solution was shown to give little or no isotopic separation. Column studies of the carbonate system exchange reaction were concluded with a 40°C run. Slightly higher enrichment of N^{15} was obtained than at 30°C. The temperature dependence of α_{eff} in this system was measured between 15 and 45°C. The factor increases with temperature, showing a tendency toward a maximum near 45°C. Isotopic exchange was found to occur between ammonia gas and aqueous carbamate ion. The exchange appears to be complete in less than 3 min. A qualitative examination was made of the carbonate system waste reflux reaction in laboratory equipment. No insurmountable difficulties are anticipated in connection with this reaction. The critical product-reflux reaction is being studied in pilot-scale equipment. Preliminary data are encouraging. Additional N exchange reactions have been studied to provide a broader basis for selecting a system for large-scale production of enriched N isotopes. A proposed system for enriching K isotopes was found to have a single-stage separation factor of $\alpha_{(30°C.L.)} = 1.006 \pm 0.002$. The single-stage fractionation factor between uranyl ion on Dowex 50 resin and on aqueous solution of uranyl ion in the presence of Versene was found to be approximately 1.00006. Very little enrichment resulted from passing a band of uranyl ion down an ion exchange column. (auth)

MATHEMATICS

8558 CF-56-3-61
Oak Ridge National Lab., Tenn.
A NEW APPROACH TO COMPUTER DESIGN. A. C. Downing, Jr. Mar. 15, 1956. 10p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The possibility that some kind of special-purpose counter could be inserted between the arithmetic unit and the memory is discussed. This counter would be used to improve the handling of the flow of information between the two units. (B.J.H.)

8559 CF-56-4-189
Oak Ridge National Lab., Tenn.
THERMAL EXPANSION STRESS IN CLAD SPHERICAL SHELLS. R. F. Hughes. Apr. 19, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An equation is derived for thermal expansion stresses in spherical shells clad with a dissimilar material. A graphical representation of the equation is presented for a carbon steel shell clad with a stainless steel liner at a temperature of 600°F and an internal pressure of 2000 psia. (B.J.H.)

8560 CF-56-4-202
Oak Ridge National Lab., Tenn.
STRESSES IN SPHERICAL SHELLS RESULTING FROM LINEAR HEATING OF INSIDE WALL. R. F. Hughes. April 30, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Stresses in spherical shells have been analyzed for the case of the inner surface being heated at a constant rate and no heat transfer through the outer surface. (auth)

8561 ORNL-1751
Oak Ridge National Lab., Tenn.
MATHEMATICS PANEL SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1954. W. C. Sangren, ed. Sept. 1, 1954. Decl. Mar. 5, 1957. 39p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

Activities of the Mathematics Panel are outlined in topical form. Test routines and new coding techniques were developed for the ORACLE. An analysis of leukemia incidence by x-ray exposure in mice is presented in addition to discussions of other projects in nuclear physics, reactor engineering, and biology. (K.S.)

NEUTRONS

8562 ANL-4798(Del.)
Argonne National Lab., Lemont, Ill.
PHYSICS DIVISION QUARTERLY REPORT FOR DECEMBER 1951, JANUARY AND FEBRUARY 1952. Mar. 15, 1952. Decl. with deletions Feb. 12, 1957. 80p. Contract W-31-109-eng-38. $12.30(ph OTS); $4.50(mf OTS).

The scattering of neutrons has been measured at five angles at four energies for U, Zr, Fe, and Cu, all relative to the scattering by carbon at the same angle and energy. The neutron source was the Li(p,n) reaction; the scattering and detecting setup is shown. New values for neutron resonances in As^{75}, U^{233}, normal U, and normal Zr and neutron transmission cross sections of Al^{27}, Fe, Ni^{58}, Zr, and Bi^{209} are tabulated. A study was made of liquid scintillation counters for neutrons using methyl borate-terphenyl in phenylcyclohexane mixtures. The theoretical aspect of the property of boron esters in such counters is given. The Xe content of the atmosphere was found, by isotopic dilution, to be 0.109 ppm (pressure). The atomic magnetic moment orientation, as found by neutron diffraction, is reported for the antiferromagnetic CrSb. The crystal structure of Np

metal, found from x-ray powder photographs is given in detail and interatomic distances are reported for NpC, Np_2C_3, NpC_2, HBO_2, and H_3BO_3. The fast-fission factor E, the probability of resonance escape p, the thermal utilization factor f, and the multiplication constant k are calculated for a one-dimensional lattice of natural U and H_2O. The critical properties of spherical water-boiler reactors, for use as low-level neutron sources, are discussed; values are tabulated for a core of H_2O-U^{235} and a reflector of BeO, graphite, or a mixture. An attempt is made to explain the unexpected critical mass of the Experimental Breeder Reactor. Cross sections for neutron inelastic scattering by middle-weight nuclei are calculated and plotted vs. neutron energy. The order of energy levels and the spin of K^{40} and Cl^{36} nuclei are investigated, using jj-coupled wave functions. Finally, theoretical consideration is given to the interaction between the configurations d^{n-5} sp and $d^{n-1}p$ (iron group), a direct method for determining the transformation matrix for the Dirac equation, and an interative formula for approximating an eigenvalue of a linear 2nd-order ordinary differential equation. (auth)

8563 CF-56-1-141
Oak Ridge National Lab., Tenn.
PROBLEMS OF NEUTRON POPULATION IN LOCALIZED ABSORBERS. D. K. Holmes. Jan. 27, 1956. 31p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Calculations are made of the distribution of one-velocity neutrons in absorbing regions in a uniform isotropic neutron flux field. Calculations are made according to the diffusion model and the first-flight transport model for a solid sphere, an infinite slab, a spherical shell, and an infinite solid cylinder. Calculations are also included on the flux distribution in the vicinity of localized absorbers in diffusing media. (B.J.H.)

8564 CF-1875
Clinton Labs., Oak Ridge, Tenn.
THE AGE OF FISSION NEUTRONS IN URANIUM WATER LATTICES. Harry Soodak and Stanley M. Forman. Aug. 4, 1944. Decl. Mar. 12, 1957. 10p. (A-2686). $3.30 (ph OTS); $2.40(mf OTS).

The age of fission neutrons in uranium–water lattices of various volume ratios is calculated taking into account the slowing down of neutrons by inelastic collisions with U. It is found that the contribution of the inelastic scattering in U to the slowing down is very appreciable. (auth)

8565 LRL-110
California Research and Development Co., Livermore Research Lab., Livermore, Calif.
NEUTRON PRODUCTIONS FROM FINITE TARGETS. C. C. Old. Apr. 1954. Decl. Feb. 26, 1957. 21p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

The neutron production in an infinite uranium MTA target bombarded by high energy deuterons was calculated. The limitations of such a calculation are reviewed, and a program for treating semi-infinite targets is developed. The various sources of neutrons from the deuteron interactions are determined. These are then used in a one-energy group diffusion calculation to predict neutron production/deuteron ratio. (auth)

NUCLEAR PHYSICS

8566 LWS-24730
California Research and Development Co., Livermore, Calif.
NEUTRON SPECTRA FROM A URANIUM TARGET BOMBARDED WITH 190 MEV DEUTERONS. A. V. Shelton. Feb. 5, 1953. Decl. Feb. 27, 1957. 12p. (CRD-R-35). $3.30(ph OTS); $2.40(mf OTS).

The spectrum of neutrons emitted by a 0.25 in. uranium target when bombarded by 190 Mev deuterons was measured. The results are not absolute, but are useful as indications of the relative yields in terms of energy, at several angles. The energies measured extend from 20 Mev to 170 Mev. A four chamber proportional counter telescope was used in the investigation. Data are presented graphically. (auth)

8567 UCRL-2285
California. Univ., Berkeley. Radiation Lab.
ENERGY SPECTRUM OF 320 MEV DEUTERONS FROM He^3. John Ise, Jr. July 17, 1953. Decl. Mar. 6, 1957. 7p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80 (mf OTS).

The energy spectrum of the deuteron beam from the stripping of He^3 was recalculated using a value of 3.75 barns for the attenuation cross section of deuterons in U. The results are presented in graphical form. (B.J.H.)

8568 UCRL-2512
California. Univ., Berkeley. Radiation Lab.
MONTHLY PROGRESS REPORT NO. 130 [FOR] JANUARY 15, 1954 TO FEBRUARY 15, 1954. Mar. 9, 1954. Decl. Mar. 4, 1957. 17p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Progress notes on experimental and theoretical physics, MTA target studies, accelerator construction and operation, chemistry, biology and medicine, and plant and equipment development are included. (D.E.B.)

NUCLEAR PROPERTIES

8569 BNL-1573
Brookhaven National Lab., Upton, N. Y.
CALCULATION OF TRANSPORT CROSS-SECTIONS OF URANIUM. S. Oleksa. Jan. 9, 1953. Decl. Feb. 14, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

8570 CF-51-8-102
Oak Ridge National Lab., Tenn.
ANALYSIS OF LID TANK NEUTRON DATA FOR BORON CARBIDE (B_4C). R. Zirkind. Aug. 2, 1951. Decl. Feb. 14, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

An analysis of the Lid Tank neutron data for B_4C-H_2O is made. An "effective removal" cross section for fast neutrons in B_4C is obtained. This is based on the "one collision theory" of shielding. The value of about 0.9 barns is obtained by the use of methods previously used for the analysis of the iron-water and lead-water Lid Tank data. (auth)

8571 CF-56-1-40
Oak Ridge National Lab., Tenn.
NEUTRON ACTIVATION OF IODINE NEAR 25 KEV. R. L. Macklin. Jan. 6, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A value of 0.78 ± 0.06 barns was found for the activation cross section near 25 kev using an antimony-beryllium neutron source. (auth)

8572 HW-23729
[Hanford Works, Richland, Wash.].
REVIEW OF THE XENON PROBLEM. P. F. Gast. Mar. 12, 1952. Decl. Feb. 20, 1957. 25p. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Two experiments have been carried out with Hanford reactors to determine the nuclear constants of xenon. These were of the type in which the reactivity following shutdown of a reactor is determined as a function of the time. It is

shown that such an experiment can determine the product of the fission yield of iodine 135 with the ratio of the xenon cross section to the fission cross section for the neutron spectrum in the shut down reactor. In symbols, (1-s) $Y\sigma_x/\sigma_f$. The values obtained from the two experiments differed by 60 per cent. One of the values is in approximate agreement with neutron spectrometer values for the cross sections and chemical values for the yield. An experiment of this type also gives a numerical value for an effective average over the pile of σ_x/σ_f for the neutron spectrum existing during the operation prior to shut down. This ratio is not identical to the one appearing in the product above and the separate factors cannot be determined by the experiment. A set of experiments is proposed in which the reactivity of freshly discharged slugs is followed in the Test Pile. This should lead to values of σ_x/σ_f as a function of moderator temperature and a reliable value of (1-s) $Y\sigma_x/\sigma_f$ for a moderator temperature of ≈20°C. Further theoretical studies will be required. Additional determinations of the amount of xenon formed directly in fission would be desirable. (auth)

8573 HW-31862

General Electric Co. Hanford Atomic Products Operation, Richland, Wash. RESONANCE CAPTURE IN THORIUM AND THORIUM OXIDE SLUGS. R. K. Cole. Aug. 30, 1954. Decl. Feb. 27, 1957. 29p. Contract W-31-109-Eng-52. $0.35(OTS).

A method is developed which relates the resonance escape probability with the parameters γ, the resonance blackness constant of a process tube; θ_0, the neutron age to resonance capture; and b, the lattice spacing. Also a method is described for measuring γ and θ_0 in the Hanford sigma pile. This method requires only a few slugs and a source of fast neutrons. These two parameters were then measured for Th slugs and ThO_2 slugs with the result that the resonance escape probabilities for a Th metal and a ThO_2 slug in a Hanford-type lattice differ by less than 2%. (auth)

8574 ORNL-1458

Oak Ridge National Lab., Tenn.
REPORT ON THE RATIO OF THE FISSION CROSS SECTION OF URANIUM-238 TO THAT OF URANIUM-235. R. R. Carlson. Feb. 19, 1953. Decl. Mar. 7, 1957. 9p. Contract W-7405-eng-26. $0.25(OTS).

The ratio of the neutron fission cross section of U^{238} to that of U^{235} has been measured at various neutron energies by means of equipment employing two parallel-plate ionization chambers with one common plate holding foils of U^{238} and U^{235} back to back and biased at -300 volts. Ionization caused by fission particles was detected by electron collection on the two outer plates. The values indicate a threshold at ~1.0 Mev and a plateau running from ~2.0 Mev to the highest neutron energy used. There is no evidence of any structure in the dependence of the ratio on energy as was observed in the U^{234}/U^{236} and U^{234}/U^{235} ratios. (L.M.T.)

NUCLEAR REACTORS

8575 ANL-WHZ-122

Argonne National Lab., Lemont, Ill.
[HANDBOOK OF CONSTANTS FOR SELECTED PILE MATERIALS.] Oct. 30, 1947. Decl. Feb. 13, 1957. 86p. Contract W-31-109-eng-38. $13.80(ph OTS); $4.80(mf OTS).

The physical and nuclear properties for a large number of reactor fuel, coolant, and structural materials are given. (D.E.B.)

8576 BMI-873

Battelle Memorial Inst., Columbus, Ohio.
THE EFFECTIVENESS OF SPRAY COOLING. R. W. Dayton, C. M. Allen, and N. E. Miller. Oct. 1, 1953. Decl. Feb. 14, 1957. 11p. Contract W-7405-eng-92. $0.25(OTS).

A possible method of cooling a liquid-fuel reactor is by spraying liquid metal through the liquid fuel, and then circulating the liquid metal through a heat exchanger. To evaluate the effectiveness of this cooling method, a few simple experiments were made with mercury sprayed through water. On the basis of the results, it was concluded that this method was intrinsically a low-power-density method, which could not find application except where a low fissionable-material inventory was the dominating requirement in a low-power reactor. Even there, it is thought that a boiling homogeneous reactor might be superior. The results are reported, in spite of their probable lack of value in the reactor program, simply to make the record complete. (auth)

8577 BNL-46

Brookhaven National Lab., Upton, N. Y.
SUPPLEMENTARY ENGINEERING TESTS FOR THE BROOKHAVEN REACTOR. D. W. Bareis and W. R. Page. Nov. 15, 1950. Decl. Feb. 14, 1957. 114p. $0.55(OTS).

An extensive thermal analysis has been made of the structure of the BNL reactor. Use was made of the electric analog technique to obtain detailed temperature patterns for various parts of the reactor structure for a number of different operating conditions. Structural changes recommended on the basis of the temperature information thus obtained were incorporated into the analogs to test the effectiveness of the changes. The analog data for both the original and final designs are presented along with a description of two auxiliary engineering tests which were run to determine air flow patterns over portions of the reactor structure where the amount of forced convection heat transfer would otherwise have been an uncertain factor. Detailed calculations of surface heat transfer coefficients simulated in the analogs are presented in appendices. (auth)

8578 BNL-111

Brookhaven National Lab., Upton, N. Y.
PRELIMINARY STUDY OF A URANIUM–BISMUTH LIQUID FUEL POWER REACTOR LFR-2. D. H. Gurinsky, I. Kaplan, F. T. Miles, C. Williams, and W. E. Winsche. June 5, 1951. Decl. Feb. 13, 1957. 33p. $0.30(OTS).

This report supersedes BNL-105.

After listing the advantages and disadvantages of a reactor using as fuel a U–Bi solution to be processed by contact with fused salts, the report presents preliminary design calculations on chemical processing of the core, construction materials, (Be, Ni, and steels of various compositions), values of core parameters, neutron economy, and a process for U^{233} production. Proposed development work on materials testing, chemical processing, fluid flow, heat transfer, small-scale experiments, and slurries as alternate fuel systems is summarized. (G.Y.)

8579 BNL-255

Brookhaven National Lab., Upton, N. Y.
EVALUATION OF BNL PILE GRAPHITE. W. L. Kosibe, D. H. Gurinsky, and G. J. Dienes. Oct. 5, 1953. Decl. Feb. 13, 1957. 18p. $0.25(OTS).

The graphite in the Brookhaven Reactor was evaluated with respect to the following properties: stored energy, physical growth, and c-axis change. As a result of these studies, it is believed that: the temperature of the graphite

in the BNL pile may be raised to 250°C or even 400°C without an excessive temperature excursion; the low temperature stored energy in the BNL pile graphite may be safely released; some growth recovery should be expected; and it is desirable to anneal the BNL pile. (auth)

8580 BNL-2097
Brookhaven National Lab., Upton, N. Y.
BOILING BURNOUT NEWSLETTER NO. 1. Corwin L. Rickard. Dec. 1, 1954. Decl. Feb. 14, 1957. 8p. $1.80 (ph OTS); $1.80(mf OTS).

New data for subcooled local boiling and quality burnout data in vertical and inclined Ni tubing are presented. Inclining the test channel apparently has little effect on the burnout point. (K.S.)

8581 CF-50-4-124(Vol.III)(Del.)
Oak Ridge National Lab., Tenn.
PILE TECHNOLOGY LECTURE NOTES. VOLUME III. Gale Young—R. C. Mason, comp. April 1950. Decl. with deletions Feb. 14, 1957. 212p. Contract [W-7405-eng-26]. $33.30(ph OTS); $9.60(mf OTS).

Lecture notes on early work in reactor technology are presented. Preliminary investigations and calculations on applications of UF_6 in reactors, reactor control, heat transfer, homogeneous reactors, and economic aspects of atomic energy are discussed. (T.R.H.)

8582 CF-50-9-38
Oak Ridge National Lab., Tenn.
RESTARTING OF HRE SOUP PUMP—SAFETY CONSIDERATIONS. W. L. Ross. Sept. 11, 1950. Decl. Feb. 16, 1957. 2p. $1.80(ph OTS); $1.80(mf OTS).

Safety studies for a soup recirculating pump show that the disc could not rupture or disrupt the system in the homogeneous reactor. An automatic interlock to stop the pump when pressure drops off in the boiler and to keep it from starting unless one safety plate is in cocked position is suggested to insure extra safety. (F.S.)

8583 CF-51-1-20
Oak Ridge National Lab., Tenn.
AN ANALYSIS OF A STEAM POWER CYCLE AS APPLIED TO A LARGE HOMOGENEOUS CIRCULATING FUEL REACTOR. J. S. Leonard and I. H. Mandil. January 11, 1951. Decl. Feb. 14, 1957. 32p. $0.35(OTS).

The cycle characteristics of steam-generating equipment, as applied to a large homogeneous circulating-fuel reactor, are investigated over a saturated steam temperature range from 100 to 580°F and a reactor fuel solution temperature range from 150 to 300°C. The reactor fuel is assumed to consist of a solution having the physical properties of ordinary water and circulated at a fixed rate of 100,000 gpm. The designed reactor power is taken at 10^6 kw with the fuel solution at 250°C and pressurized to 1000 psi. Saturated steam at 200 psia is generated in a submerged-tube steam generator with the heating surface based on an over-all heat-transfer coefficient of 1000 Btu/hr/ft^2/°F and utilizing $\frac{3}{4}$-in. OD by 0.277-in. ID stainless steel tubes. Vapor-binding characteristics are investigated. Data are presented indicating the effects of steam generator tube size and flow velocity variations on heat-transfer coefficients, pressure drop across the tubes, and solution holdup in the tubes. (C.H.)

8584 CF-51-7-132
Oak Ridge National Lab., Tenn.
DECREASE IN PILE REACTIVITY DUE TO BUILD-UP OF FISSION PRODUCTS. D. T. Bray. July 27, 1951. Decl. Mar. 11, 1957. 8p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Assuming removal of all the high cross section fission product gases such as Xe, the decrease in pile reactivity due to fission product build-up is plotted for homogeneous reactors. (D.E.B.)

8585 CF-51-11-94
Oak Ridge National Lab., Tenn.
NUCLEAR CONSTANTS OF THE HRE. H. T. Williams. Nov. 19, 1951. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Results of calculations of the nuclear constants of the HRE are briefly summarized. (B.J.H.)

8586 CF-51-12-119
[Oak Ridge National Lab., Tenn.]
PRELIMINARY INVESTIGATION OF PIPE GAS SEPARATORS BY J. I. GONZALES. I. Spiewak. Dec. 10, 1951. Decl. Feb. 15, 1957. 10p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8587 CF-51-12-122
Oak Ridge National Lab., Tenn.
PHOTONEUTRONS IN THE HRE. H. T. Williams. Dec. 19, 1951. Decl. Mar. 12, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Estimates are made of the effect of the photoneutrons from the γ-n reaction in D_2O on the kinetic behavior of the HRE. (B.J.H.)

8588 CF-52-1-76(Del.)
[Oak Ridge National Lab., Tenn.]
HEAT TRANSFER IN NUCLEAR REACTORS. R. N. Lyon and W. B. Harrison. [Jan. 1952]. Decl. with deletions Feb. 14, 1957. 59p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60(mf OTS).

The theory of heat generation in nuclear reactors is discussed. Types of heat flow paths for use in cooling nuclear reactors and methods of predicting temperatures along these paths are summarized. The various heat transfer features of a number of reactors in operation are presented. Emphasis is directed toward typical applications of heat-transfer principles to general reactor design. (W.L.H.)

8589 CF-52-8-161(Del.)
Oak Ridge National Lab., Tenn.
ENRICHMENT, PLUTONIUM PRODUCTION RATE AND SLURRY COSTS FOR D_2O-UO_3 AND H_2O-UO_3 SLURRY REACTORS AT 250°C. Clayton D. Zerby. Aug. 22, 1952. Decl. with deletions Feb. 14, 1957. 12p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8590 CF-52-10-216
Oak Ridge National Lab., Tenn.
ESTIMATE OF THE EFFECT OF POISONS ON CRITICAL CONCENTRATION. R. E. Aven, P. N. Haubenreich, and M. Tobias. Oct. 30, 1952. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

If a highly-enriched reactor is operated without removal of fission products, these poisons will build up so that it is necessary to increase the fuel concentration to maintain criticality at the original operating temperature. Thus the fuel which must be added over a period of time is greater than that burned up by the amount necessary to increase the concentration. To determine the magnitude of this concentration change, calculations have been made for a highly-enriched reactor operating at 50 megawatts. These calculations indicate that after the first year of operation the necessary U^{235} concentration for criticality is 1.2 times the original concentration, and after five years, about twice the original concentration is required. The exact value for the concentration depends upon the assumed values of poison cross sections. (auth)

8591 CF-53-1-94
Oak Ridge National Lab., Tenn.
HRE OPERATIONS MANUAL. T. H. Thomas, ed. Oct. 9, 1952. Decl. Feb. 16, 1957. 56p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60(mf OTS).

Personnel job descriptions and detailed start-up, operating, and shut-down instructions for the HRE are given. (D.E.B.)

8592 CF-53-1-110
Oak Ridge National Lab., Tenn.
GAS PRODUCTION IN THE ISHR. J. P. Sanders. Jan. 6, 1953. Decl. Feb. 14, 1957. 10p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The rate of formation of D_2 and O_2 in the ISHR is calculated and the composition of the evolved gas is discussed along with partial pressures in the gas separator. Steady state concentrations of gases dissolved in the liquid entering and leaving the core are calculated using the assumption that the gas in the bubbles is in equilibrium with the liquid and that all the undissolved gas is removed in the separator. (T.R.H.)

8593 CF-53-1-323
Oak Ridge National Lab., Tenn.
CONVERSION RATIO AS A FUNCTION OF RADIUS IN A THORIUM BREEDER. M. Tobias. Jan. 29, 1953. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Calculations made to determine conversion ratio as a function of radius in the blanket of the ISHR are given. It was found that conversion was about 91% completed by absorptions occurring in the first foot of blanket thickness in the most dilute blanket considered. For the two more concentrated cases studied, conversion was 95% or more completed in the first foot. Graphs and tables are presented showing the variation in detail. (auth)

8594 CF-53-5-112
Oak Ridge National Lab., Tenn.
TWO-PHASE FLOW. J. A. Hafford. May 18, 1953. Decl. Feb. 13, 1957. 11p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Design criteria for the ISHR gas separator and for future two-phase flow problems are summarized. (D.E.B.)

8595 CF-53-5-119
Oak Ridge National Lab., Tenn.
A TWO-GROUP TWO-REGION APPROXIMATION FOR BLANKET BREEDER REACTORS. M. C. Edlund. May 19, 1953. Decl. Feb. 7, 1957. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

A simple two-group model which is equivalent to a zero order Feynman-Welton approximation, a recipe for calculation of critical mass and breeding ratio, and a comparison of these results with those obtained from the usual two-group elementary diffusion theory are discussed. (auth)

8596 CF-53-5-208
Oak Ridge National Lab., Tenn.
HRE POWER AND TEMPERATURE AFTER STOPPING PUMP. Sidney Visner. May 28, 1953. Decl. Feb. 14, 1957. 9p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

There has been considerable interest in the expected temperature rise in the HRE core in the event the main circulating pump stopped with the reactor power at 1 megawatt. The core temperature cannot be measured reliably with the pump stopped, although the total pressure can be observed. It is conceivable that the temperature rise from the power generated in core due both to radioactive decay of the fission products and to continued fissioning may result in excessive pressures. Although the present control system for the reactor requires the fuel solution to dump on failure of the 100 A pump, dump tests have revealed that approximately 30 secs are required after the dump valve is opened for the liquid level to fall below the top of the core and of the order of 5 minutes to empty an appreciable fraction of the core. In view of the fact that the rate of temperature rise is 5.7°C per sec at a power input of 1 megawatt, it is conceivable that an unsafe condition may be reached. On the other hand, if the temperature rise is not excessive, it may not be necessary to dump the fuel due to a minor interruption of the pump operation; thus avoiding the subsequent start-up procedure. When operating with a core outlet temperature of 250°C, the present calculations indicate that the temperature will not exceed 282°C after the circulating pump stops. Since the corresponding pressure is 940 psig, the situation should not be dangerous. The time dependence of the power and temperature for the first 30 secs is also obtained. There is also the hazard to be considered of re-starting the circulating pump after the heat exchanger has been permitted to cool appreciably and in this manner introducing reactivity very rapidly into the reactor. A final evaluation of this hazard will be made from tests on the reactor. (auth)

8597 CF-53-6-158
Oak Ridge National Lab., Tenn.
THE INFLUENCE OF THE METHOD OF CALCULATING RESONANCE CAPTURE ON THE FAST LEAKAGE IN BREEDERS. M. Tobias and M. C. Edlund. June 16, 1953. Decl. Mar. 12, 1957. 21p. Contract [W-7405-eng-26]. $0.35(OTS).

8598 CF-53-8-95(Del.)
Oak Ridge National Lab., Tenn.
STATUS OF THORIUM BLANKET DEVELOPMENT. H. F. McDuffie. Aug. 14, 1953. Decl. with deletions Feb. 14, 1957. 27p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Thorium slurries, solutions, and solids have been proposed for blanket applications in homogeneous reactors. These types were evaluated on the basis of their known characteristics, and areas demanding further study are reviewed. From the information available, a ThO_2 circulating slurry appears most desirable for a homogeneous power breeder. Pump wear is seen as a prime source of concern. (D.E.B.)

8599 CF-53-9-30
Oak Ridge National Lab., Tenn.
VARIATION OF BREEDING RATIO WITH THORIUM BLANKET CONCENTRATION IN A TWO-REGION THERMAL BREEDER. Melvin Tobias. Sept. 3, 1953. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Calculations using the adaptation of the Feynmann-Welton method described by M. C. Edlund were performed to estimate the variation of breeding ratio with blanket concentration. The reactor configuration used was a six-foot-diameter spherical core surrounded by a two-foot-thick blanket. The two regions were assumed separated by a one-half-inch thick Zr shell. The blanket concentrations considered were 4218, 2000, 1000, 500, and 250 gm Th per liter of blanket volume. The moderator in both core and blanket was D_2O, and the fuel compound was UO_2SO_4. The results indicated a slow decline in breeding ratio from 1.16 at 4218 gm Th/liter to 1.04 at 500. This suggests that there may be a considerable economic advantage to trading a small loss in breeding gain for more rapid blanket turnover. (auth)

8600 CF-54-2-158
Oak Ridge National Lab., Tenn.
HRT DESIGN DATA. F. C. Zapp and R. E. Aven. Feb. 22, 1954. Decl. Feb. 15, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

General reactor data for the core solution, main fuel loop and components, heat exchangers, main blanket loop and components, pumping requirements, and gas separator are tabulated for the Homogeneous Reactor Test. The reactor has an over-all gross power rating of 6.6 mw and is a two-region converter for the production of low g/T Pu. (F.S.)

8601 CF-54-2-208
Oak Ridge National Lab., Tenn.
POISONING OF THE CORE OF THE TWO-REGION HOMOGENEOUS THERMAL BREEDER: STUDY NO. 2. A. T. Gresky and E. D. Arnold. Feb. 22, 1954. Decl. Feb. 15, 1957. 26p. Contract [W-7405-Eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Further evaluation of the magnitude and nature of fission and corrosion product poisoning of the core solution in the Two-Region Homogeneous Thermal Breeder has been made. This study indicates both graphically and in tables the contribution of the various poisons over periods of 40 to 150 days irradiation. There are nineteen isotopes, comprising a yield of 47%, that add considerable poison to the core system. Approximately 72% of the known and certain fission product poisons are contributed by the isotopes of the rare earth group. However, there are two uncertain poisons (Tc^{99} and Sr^{89}) which have high, but of questionable accuracy, cross-sections. When these are included in the total poison contribution, then the rare earth group contributes only 55% of the total fission product poison. These two "uncertains" were included in order to give as complete an analysis as possible and also to take a more pessimistic view of poison effects. Poison due to corrosion products was also found to be a major problem. It has been determined that if corrosion rates were as high as 1.5 mils per year, corrosion product poisons would become equal in magnitude to that contributed by the non-rare earth groups. The probable chemical behavior of the various chemical groups was discussed with special emphasis on strontium and technetium. (auth)

8602 CF-54-3-77
Oak Ridge National Lab., Tenn.
FURTHER CALCULATIONS FOR URANIUM-FUELED REACTORS. Supplement to CF-53-12-1. P. N. Haubenreich. Mar. 16, 1954. Decl. Feb. 15, 1957. 6p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The calculations described in ORNL-CF-53-12-1 for U-fueled reactors are recalculated using revised values for the nuclear properties of Pu. (B.J.H.)

8603 CF-54-3-142
Oak Ridge National Lab., Tenn.
FLUX DISTRIBUTION IN THE HRT WITH 300 g/LITER NATURAL URANIUM BLANKET. Phillip M. Wood. Mar. 22, 1954. Decl. Feb. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Two-group diffusion theory was used to calculate neutron flux distributions and adjoint functions in the Homogeneous Reactor Test. Calculations were made for the case of a 300 g/l natural uranium blanket and a 93.5% enriched UO_2SO_4 core. Flux and adjoint functions are shown as a function of distance from the center of the core, and heat production in the blanket is shown as a function of distance from the core. (B.J.H.)

8604 CF-54-4-105
Oak Ridge National Lab., Tenn.
HRT DESIGN REQUEST NO. 2. TITANIUM COMPONENTS FOR ALTERNATE BLANKET SYSTEM. R. B. Briggs, J. W. Hill, and J. R. McWherter. Apr. 15, 1954. Decl. Feb. 15, 1957. 11p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Design data for Ti blankets, gas systems, heat exchangers, and circulation systems for the Homogeneous Reactor Test are given. (F.S.)

8605 CF-54-4-112
Oak Ridge National Lab., Tenn.
POSSIBILITY OF BRITTLE PRESSURE VESSEL FAILURE. E. C. Miller. Apr. 16, 1954. Decl. Feb. 7, 1957. Includes memo IMPACT STRENGTH OF ASTM-212 STEEL. J. C. Wilson. Apr. 9, 1954. 9p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

In connection with HRT design, the history of brittle failures of welded engineering structures is reviewed. Available data on the impact strength of ASTM-212 steel are included. (B.J.H.)

8606 CF-54-5-26
Oak Ridge National Lab., Tenn.
ESTIMATION OF MOST ECONOMICAL VELOCITY IN REACTOR CIRCULATING SYSTEMS. P. N. Haubenreich. May 5, 1954. Decl. Feb. 15, 1957. 18p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8607 CF-54-5-46
Oak Ridge National Lab., Tenn.
HEAT PRODUCTION IN HRT SHIELD AND ARGON ACTIVATION IN REACTOR AND EQUIPMENT CELLS. R. B. Briggs. May 7, 1954. Decl. Feb. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Neutrons which leak from the reactor and the nuclear properties of shield materials are mathematically analyzed. It is found that there should be a 2-foot inner shield around the reactor to reduce heat production in the main shield walls and activation of A. If the inner shield is borated it will be unnecessary to include B in the pool walls. (F.S.)

8608 CF-54-5-100
Oak Ridge National Lab., Tenn.
ESTIMATE OF O_2 GENERATED FROM RADIOLYTIC DECOMPOSITION OF BOILER WATER IN THE FUEL SOLUTION HEAT EXCHANGER OF THE HRT AT 5 Mw OPERATION C. L. Segaser. May 18, 1954. Decl. Feb. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8609 CF-54-5-165
Oak Ridge National Lab., Tenn.
LOSS OF DELAYED NEUTRONS IN HOMOGENEOUS, BOILING REACTORS. P. R. Kasten. May 19, 1954. Decl. Feb. 15, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A study has been made of the delayed neutron losses which could occur as a result of steam vapor stripping the delayed neutron precursors from a boiling reactor. No detrimental effects due to loss of delayed neutrons are foreseen. Operating conditions considered were as follows: power densities from 5 to 10 kw/liter of core fluid, pressures from 600 to 2000 psia, fraction vapor in the core from 10 to 30%, and the steam chest volume twice that of the reactor core. Under the above conditions, roughly one-third of the delayed neutrons can be lost from the core as a result of steam stripping of the delayed neutron precursors. Typical values are as follows: for U^{235}, β, the effective fraction of delayed neutrons, decreases from 0.00755 to 0.00572; for U^{233}, β decreases from 0.00242 to 0.00176; and for Pu^{239}, β decreases from 0.00364 to 0.00253. (auth)

8610 CF-54-7-26
Oak Ridge National Lab., Tenn.
USE OF THE HRE CHARCOAL ABSORBERS IN THE HRT. I. Spiewak. July 8, 1954. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Pressure drop and Xe and Kr holdup through the HRE charcoal absorbers were evaluated to ascertain the possibility of their use for the HRT. Satisfactory performance is indicated. (D.E.B.)

8611 CF-54-7-125
Oak Ridge National Lab., Tenn.
DELAYED NEUTRON INTENSITIES IN HRT AND K-49 CORE SOLUTIONS. E. D. Arnold. July 16, 1954. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Analyses have been performed to determine the intensity of delayed neutrons in core loop solutions for the HRT and K-49 reactors, to establish the effect of decay time on the intensity, and to determine the extent of the activity created by capture of the delayed neutrons during steady-state operation and after shutdown. Numerical results are presented. (B.J.H.)

8612 CF-54-7-202
Oak Ridge National Lab., Tenn.
HRT NUCLEAR INSTRUMENTATION PROPOSAL. J. N. Baird. July 29, 1954. Decl. Feb. 15, 1957. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The HRT instrumentation program is briefly outlined, including plans for operational equipment, experimental equipment, personnel monitoring equipment, and installation proposals. (B.J.H.)

8613 CF-54-8-53
Oak Ridge National Lab., Tenn.
HRT DESIGN DATA, ISSUE NO. 2. F. C. Zapp and R. E. Aven. Aug. 10, 1954. Decl. Feb. 15, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Design data sheets for a two-region power producer having an over-all gross design power rating of 5220 kw and using a D_2O blanket solution are presented. (F.S.)

8614 CF-54-8-61
[Oak Ridge National Lab., Tenn.]
PRELIMINARY NUCLEAR CALCULATIONS FOR THE THORIUM BREEDER REACTOR. P. H. Haubenreich. Aug. 13, 1954. Decl. Feb. 15, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Preliminary two-region calculations for the Thorium Breeder Reactor have been done on the ORACLE. Calculations for reactors having 4, 5, and 6-ft. cores, blanket Th concentrations of 500 and 1000 g/l and blanket U^{233} concentrations of 0, 3, and 5 g/kg Th are given. (D.E.B.)

8615 CF-54-8-214
Oak Ridge National Lab., Tenn.
HRP–CP: SUMMARY OF DESIGN CRITERIA AND PURPOSES OF LOOP 3–SOLID-LIQUID SEPARATOR LOOP. W. L. Carter. Aug. 31, 1954. Decl. Feb. 15, 1957. 11p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The design of a test loop for studies of fission product separation from HRT fuel solutions is discussed. Fission product concentrations as a function of reactor and cyclone separator operating times are plotted. (D.E.B.)

8616 CF-54-9-64
Oak Ridge National Lab., Tenn.
HRT POWER AND FUEL TEMPERATURE AFTER FUEL CIRCULATION STOPS. J. W. Hill, Jr. Sept. 9, 1954. Decl. Feb. 14, 1957. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

An estimate was made of the Homogeneous Reactor Test power level and fuel temperature as a function of time after the fuel circulating pump stops, using the assumptions that the fuel is not dumped and that there is no cooling of the fuel. Data are presented graphically on the power in the core at various times after shutdown for several operating power levels and periods of operation; the total energy released after shutdown; and the temperature of the fuel solution in the core tank following infinite operation at various power levels. (C.H.)

8617 CF-54-9-129
Oak Ridge National Lab., Tenn.
STRAIGHT-THROUGH HRT CORE MODEL TESTS. P. H. Harley. Sept. 22, 1954. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A straight-through core suitable for HRT use has been developed. It contains nine screens in the inlet diffuser, which should be preceded by a vaned elbow. Measurements of gas holdup in both straight through and alternate HRT cores have been obtained. (auth)

8618 CF-54-10-16
Oak Ridge National Lab., Tenn.
TWO-REGION THORIUM BREEDER-POWER REACTOR. PROGRESS REPORT NO. 1. H. C. Claiborne, M. Tobias, and T. B. Fowler. Oct. 4, 1954. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

In connection with the K-23 design program, investigation of the effects of major variables on the economics and characteristics of two-region throium breeder-power reactors has been started. The entire problem has been coded for calculation by the ORACLE. The results for nearly 100 reactor calculations show that the net unit cost of electricity is only slightly affected by rather large changes in the process variables. For reasonable ranges of the major variables (core size, blanket thickness, thorium concentration, U^{233} blanket concentration, poisons) the unit cost was between 6.3 and 6.6 mills/kwh for a 3-reactor station rated at 375 Mw of electrical power and operated at 280°C. At economic optimum conditions, the core densities rise to nearly 400 kw/liter in some cases. Therefore, it is probable that operation at an economic optimum becomes limited by engineering considerations. However, this is not serious in view of the insensitiveness of the unit cost to changes in the process variables. Perhaps the most important result of this study is that increasing the thorium blanket concentration from 1000 to 1500 gm/liter reduces the unit cost by only a negligible amount (0.02 mills/kwh) and the difference between 500 and 1500 gm/liter is only 0.15 mills/kwh. Even for 500 gm/liter, the breeding ratio was greater than unity. (auth)

8619 CF-54-10-96
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROCESSING: CONSIDERATIONS IN APPLYING LIQUID-SOLID CYCLONE SEPARATORS TO HOMOGENEOUS REACTOR PROCESSING. M. E. Whatley. Oct. 20, 1954. Decl. Feb. 15, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Solid-liquid cyclone separators were studied for use in separating the precipitated solids from homogeneous reactor fuel solutions. Tests made with a commercial unit at atmospheric pressure and room temperature were encouraging. Design development studies are reported on a unit of high efficiency to operate at feed flows on the order of 0.1 gpm. Cyclone cascade combinations considered are illustrated and data on the efficiencies of cascade combinations of liquid-solid cyclones are compared with the efficiency of the unit cyclone. Performance data are discussed. (C.H.)

8620 CF-54-11-172
Oak Ridge National Lab., Tenn.
POSSIBILITY OF PEROXIDE PRECIPITATION IN HRT DUMP AND SAMPLER LINES. Phillip M. Wood. Nov. 29, 1954. Decl. Feb. 15, 1957. 6p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Peroxide decomposition and beta power density are plotted as a function of temperature. The effects of these variables on peroxide precipitation are discussed, and it is shown that there is a danger of precipitation if the lines are operated below 70°C. (D.E.B.)

8621 CF-54-11-190
Oak Ridge National Lab., Tenn.
DESIGN CRITERIA FOR THE HRT CHEMICAL PLANT. William L. Carter. Nov. 24, 1954. Decl. Feb. 15, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A detailed discussion of the design requirements as well as suggested means of accomplishing the several operations is presented. (auth)

8622 CF-54-12-34
Oak Ridge National Lab., Tenn.
HR PROCESSING DEVELOPMENT: APPLICABILITY OF THE BULK SHIELDING FACILITY REACTOR FOR THE RADIATION STUDIES REQUIRED. Paul A. Haas. Dec. 3, 1954. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Development of Homogeneous Reactor (HR) chemical processing required investigation of the effects of radiation on the material to be processed. Use of the Bulk Shielding Facility (BSF or "Swimming Pool") was proposed for these radiation studies. This report summarizes the results of this evaluation, presenting the advantages and disadvantages of using the BSF for HR chemical processing development studies. (auth)

8623 CF-55-2-82
Oak Ridge National Lab., Tenn.
HRT CP EXPERIMENTAL EVAPORATOR STUDIES—PROBLEM STATEMENT. J. C. Bresee. Feb. 11, 1955. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

To recover the D_2O from the solution removed from the HRT core, it is proposed that the solution be evaporated to ~50 wt. % UO_2SO_4 in the HRT chemical plant, thereby effecting 97% recovery of the D_2O. A study is made of the evaporation from 1.6 to 50 wt. % UO_2SO_4, including problems of entrainment and condensate contamination, evaporator control, solution removal, and corrosion. Designs for experimental evaporators are included. (B.J.H.)

8624 CF-55-6-53
Oak Ridge National Lab., Tenn.
DESIGN REVIEW COMMITTEE REPORT ON THE HOMOGENEOUS REACTOR TEST (HRT). C. J. Borkowski. June 7, 1955. Decl. Feb. 16, 1957. 32p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The HRT Design Review Committee reported on various aspects of the HRT design including, engineering design, equipment layout, piping drawings, control and instrumentation, reactor kinetics and soup chemistry. (F.S.)

8625 CF-55-6-57
Oak Ridge National Lab., Tenn.
MINUTES OF HRP GROUP LEADERS MEETING. M. G. McDuffie. June 9, 1955. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The status of HRT construction was reviewed. Slurry investigations include the effect of change in calcination temperature on the properties of thorium oxide, solid characteristics and particulate properties of various oxides, the effects of additives, and the rheologic properties of slurries. The corrosive effects of slurries are being investigated, and results are reported on corrosion loop runs. Efforts to achieve a modification of the HRT dump tanks to permit safe handling of slurries were discussed. (M.P.G.)

8626 CF-55-7-84
Oak Ridge National Lab., Tenn.
A STUDY OF DESIGN CRITERIA FOR A PROPOSED HRT HEAT EXCHANGER BLAST SHIELD. C. L. Segaser. July 18, 1955. Decl. Feb. 14, 1957. 22p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

In order to prevent the release of radioactive products during a rupture, it has been recommended that a blast shield be installed around the HRT heat exchangers. The shield should be of sufficient strength to contain both the pressure and impact forces which result from a heat exchanger failure. Complete calculations concerning such a shield are included. (B.J.H.)

8627 CF-55-8-169
Sharples Corp., Philadelphia.
FEASIBILITY STUDY OF A CANNED ROTOR CENTRIFUGE FOR USE IN HOMOGENEOUS REACTOR AND OTHER RADIOACTIVE SYSTEMS. Charles M. Ambler. Aug. 30, 1955. Decl. Mar. 12, 1957. 51p. [For Oak Ridge National Lab.]. Contract W-7405-eng-26, Subcontract 574. $0.50(OTS).

It is feasible to construct a canned rotor centrifuge for 1 year maintenance free operation at 2050 psia feed pressure and 2000 psia discharge pressure under O_2 over pressure. The centrifuge will operate at 300°C in a high radiation field with zero leakage. (T.R.H.)

8628 CF-55-9-27
Oak Ridge National Lab., Tenn.
HRT URANYL SULFATE START-UP PROCEDURE. E. L. Compere and E. G. Bohlmann. Sept. 7, 1955. Decl. Feb. 15, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Before the startup of the HRT, shakedown tests with demineralized water should be completed and then the system should be cleaned with a tri-sodium phosphate wash. Distilled water is then added, the system is brought to operating condition, and the uranyl sulfate is added. Samples should be taken twice per shift for pH, U, Ni, Cu, Cr, and solids. Daily analysis should include Zr, SO_4, Co, O_2, CO_2, and inert and residual gases. (F.S.)

8629 CF-55-10-39
Oak Ridge National Lab., Tenn.
MINUTES OF HRP GROUP LEADERS MEETING, AUGUST 18, 1955. H. F. McDuffie. Oct. 11, 1955. Decl. Feb. 23, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Brief summaries of the Homogeneous Reactor Test status and research on physical properties of ThO_2 slurries, heat transfer to slurries flowing turbulently through tubes, and reactor radiation effects on corrosion of Zircaloy-2 in uranyl sulfate solutions are given. (G.Y.)

8630 CF-55-12-149
Oak Ridge National Lab., Tenn.
CONCENTRATION OF D_2 AND O_2 IN FUEL FEED STREAM AS IT ENTERS THE HIGH-PRESSURE CIRCULATING SYSTEM OF THE HRT. R. E. Aven. Dec. 28, 1955. Decl. Feb. 16, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A suggestion has been made to introduce the fuel feed stream into the stagnant liquid leg leading to the pressurizer. Calculations are carried out which indicate the upper limit of D_2 and O_2 concentrations in the fuel stream as it enters the high-pressure circulating system. The value obtained for D_2, ~0.45 mole %, appears safe. (D.E.B.)

8631 CF-56-1-26(Del.)
Oak Ridge National Lab., Tenn.
STATUS AND OBJECTIVES—HOMOGENEOUS REACTOR PROJECT. SUMMARIES OF PRESENTATIONS TO THE REACTOR SUBCOMMITTEE OF THE GENERAL ADVISORY COMMITTEE. S. E. Beall, E. G. Bohlmann, R. B. Briggs, F. R. Bruce, J. A. Lane, R. N. Lyon, C. H. Secoy, J. A. Swartout, and C. E. Winters. Jan. 10, 1956. Decl. with deletions Feb. 15, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8632 CF-56-2-47
Oak Ridge National Lab., Tenn.
HRT FUEL TRANSFER RATES BETWEEN INNER AND OUTER DUMP TANKS. H. K. Search. Feb. 10, 1956. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Calculations were made to determine flow rates at various dump tank pressures for various conditions. The calculations indicate a possibility that a dangerous amount of fuel could be transferred to the core if the dump valve is open while transferring fuel from the outer to the inner dump tank. (B.J.H.)

8633 CF-56-2-69
Oak Ridge National Lab., Tenn.
HRT PRESSURIZER DESIGN. Paul R. Kasten. Feb. 16, 1956. Decl. Mar. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An examination was made of the present HRT pressurizer design with respect to reactor safety. Particular emphasis is placed on the frictional resistance to fluid flow between the core and pressurizer and on the volume of hot liquid within the pressurizer. (B.J.H.)

8634 CF-56-2-158
Oak Ridge National Lab., Tenn.
ORR PROJECT MONTHLY PROGRESS REPORT NO. 17. J. P. Gill and T. E. Cole. Feb. 15, 1956. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Progress in the construction of the ORR is briefly reviewed. Experiments were made in the BSF to observe the gross heating effect in the ORR. Discussions are also included on the progress in instrumentation, experimental facilities, and component design. (For preceding period see ORNL-CF-55-12-158.) (B.J.H.)

8635 CF-56-2-166
Oak Ridge National Lab., Tenn.
FIVE LECTURES ON NUCLEAR REACTORS PRESENTED AT CAL TECH. Alvin M. Weinberg. Feb. 10, 1956. 37p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

The basic issues involved in the physics and engineering of nuclear reactors are summarized. Topics discussed include theory of reactor design, technical problems in power reactors, physical problems in nuclear power production, and future developments in nuclear power. (C.H.)

8636 CF-56-3-12
Oak Ridge National Lab., Tenn.
HELIUM LEAK TEST ON HRT EXCHANGER NO. 8532 AT Y-12. S. I. Kaplan. Mar. 1, 1956. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

HRT Heat Exchanger No. 8532 was helium leak-tested at Building 9204-1, Y-12 Plant, on February 14-16, 1956. A leak rate of about 0.006 cc per day was found between the tube and shell sides when the tubes were pressurized to 1235 psia with helium and the shell evacuated to 7 μ. Sample calculations are presented for gas flows through capillary leaks. (auth)

8637 CF-56-3-13
Oak Ridge National Lab., Tenn.
EVALUATION OF HEAT LOSS AND INSULATION PERFORMANCE ON FUEL HEAT EXCHANGER DURING PRELIMINARY TEST. Work Period: February 12–13, 1956. J. D. Perret, Jr. Mar. 5, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The over-all heat loss from the heat exchanger shell to air was found to be 19,350 Btu/hr for a shell to air ΔT equal to 317°F. Taking the median heat transfer area to be equal to 117 ft^2, the over-all heat transfer coefficient through shell, insulation, blast shield, and air film is U = 0.53 Btu/hr-ft^2-°F. (auth)

8638 CF-56-3-31
Oak Ridge National Lab., Tenn.
SURFACE TEMPERATURES IN THE ORR TANK NEAR THE POOLSIDE FACILITY. E. S. Bettis and F. T. Binford. Mar. 7, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The surface temperature to be expected in the reactor tank wall in the vicinity of the poolside facility is briefly discussed. The use of lead shielding between the Al wall and the reactor to reduce the maximum heat generation is also discussed. (B.J.H.)

8639 CF-56-3-34
Oak Ridge National Lab., Tenn.
DESIGN OF SOLIDS FEED LOOP FOR SHAKEDOWN OF THE HRT CHEMICAL PLANT. B. H. Hamling. Mar. 5, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A 400-gpm pump loop will be used to supply the HRT Chemical Plant with water and simulated reactor corrosion and fission products in natural uranyl sulfate solution at 300°C and 2000 psi for shakedown purposes. The loop will have no hot metal surfaces on which the corrosion and fission products might adhere. Heating of the soup will be entirely from the brake horsepower of the pump. It is uncertain whether the loop can be heated up to 300°C by this method without causing increased wear on the graphitar bearings in the pump. A large portion of the heat available for heating up the loop is normally lost in the pump cooling water stream. Reducing this heat loss by decreasing the cooling water flow causes higher temperatures in the motor and increases bearing wear. It is recommended that the more efficient insulation, Santocel, be employed throughout to reduce heat losses and minimize the work input required from the 400A pump (this will permit running the pump bearings at lowest temperatures). (auth)

8640 CF-56-3-93
Oak Ridge National Lab., Tenn.
UNOP "B" LOOP DESIGN MODIFICATION FOR HRT-CP COMPONENT AND MOCK-UP STUDIES. P. E. Novak. Mar. 14, 1956. 13p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

From experiments conducted 2/8/56 on the B loop, approximately 20 psi was available to drive the two experiments. This pressure drop was insufficient to support the proposed tests on the existing loop but could produce satisfactory results on a modified loop design. The pressure drop available on a modified loop would produce a flow rate of 0.6 gpm through the 400 mil hydroclone test loop and 0.7 to 1.1 gpm through the flange test loop at operating temperature and pressure. A flow rate approaching 3 gpm would also be available for circulation of the underflow pot contents at room temperature. These flow rates and temperatures approach operating conditions in the HRT-CP and are acceptable for the proposed tests. The HRT-CP equipment being utilized for the initial test consists of the hydroclone and underflow pot assembly, the titanium filter screen, four electric heaters, and a 50 scfm blower. The final mock-up will also include the HRT-CP sampler and a stainless steel prototype of the canned rotor pump to be used in the chemical plant. (auth)

8641 CF-56-3-135
Oak Ridge National Lab., Tenn.
HRT HEAT EXCHANGERS—CLEANING RUNS PERFORMED AT Y-12. P. P. Holz. Mar. 20, 1956. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The HRT blanket (No. 8531) and fuel heat exchangers (No. 8532) were subjected to thorough oxygenated water flushing runs to eliminate small amounts of carbon deposits and impurities from the tube sides which had accidentally entered during thermal cycling operations at the Foster Wheeler plant. The oxygenated water flushings proved effective on both heat exchangers to assure HRT cleanliness qualifications. Utilizing the flushing loop setup, additional tests were run on the fuel heat exchanger. Uranyl sulfate solution (approx. 10 gm/l) was circulated to explore a minute leak, which was first noted during the final helium leak test of the primary side at Foster Wheeler, proved effective in reducing the leak from 0.35 to 0.50 cc/day to 0.006 cc/day. It is assumed that the leak, which was probably located in the tube joint region, was essentially sealed by corrosion products. Further, condensate sampling of the shell during the "soup" run, and analyzing the condensate for uranium, verify this conclusion. Other tests conducted in connection with the flushing loop setup included pressure drop calibrations, heat loss data tests, and a helium leak test. Specific summary reports are listed in the bibliography. (auth)

8642 CF-56-3-173
Oak Ridge National Lab., Tenn.
PULSAFEEDER AND PURGE PUMP FAILURES ON THE HRT MOCKUP. H. L. Falkenberry and I. Spiewak. March 1, 1956. 10p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Data on pulsafeeder and purge pump failures in the HRT mockup are summarized. Measures to prevent such failures in the HRT are suggested. (B.J.H.)

8643 CF-56-4-20
Oak Ridge National Lab., Tenn.
POSSIBLE SUBSTITUTES FOR FREON-11 IN HRT COLD TRAP REFRIGERATION SYSTEM. R. C. Robertson and J. E. Kuster. Apr. 5, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8644 CF-56-4-28
Oak Ridge National Lab., Tenn.
TENTATIVE TURBINE GENERATOR TEST PROCEDURE—HRT TEST V A 1. H. K. Search. Apr. 2, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8645 CF-56-4-43
Oak Ridge National Lab. Tenn.
NOTES ON HELIUM LEAK DETECTION PRACTICE FOR HRT. S. I. Kaplan. Apr. 9, 1956. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A guide is given to the equipment and methods usable for leak testing HRT components and the assembled plant. (B.J.H.)

8646 CF-56-4-65
Oak Ridge National Lab., Tenn.
BLANKET STORAGE TANKS AND REFLUX CONDENSER ITEMS 49a, b AND 39. J. L. Gory. Apr. 10, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Test procedures are given for measuring the volume of the blanket storage tanks, checking the operation of the storage tank's liquid level indicator, determining the time required to get up pressure in the storage tanks to transfer solution to the dump tanks, determining flows from the blanket storage tanks, and checking the operability and heat removal capacity of the reflux condenser. (B.J.H.)

8647 CF-56-4-103
Oak Ridge National Lab., Tenn.
REACTIVITY EFFECTS ASSOCIATED WITH TWO-PHASE SEPARATION. Paul R. Kasten. Apr. 16, 1956. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Reactivity problems associated with two-phase separation in the HRT are briefly discussed. Methods for eliminating these problems are suggested. (B.J.H.)

8648 CF-56-4-106
Oak Ridge National Lab., Tenn.
OPERATIONAL TESTS ASSOCIATED WITH USE OF FREON IN HRT SECONDARY REFRIGERATION SYSTEM. R. C. Robertson. Apr. 17, 1956. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8649 CF-56-4-115
Oak Ridge National Lab., Tenn.
BLANKET DUMP TANKS AND CONDENSATE TANKS. HRT Engineering Test Procedures No. IV 34a, b, c, and 37. W. D. Burch. Apr. 14, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Test procedures are given for calibrating the HRT dump tank weigh cell, determining the total volume of the low-pressure system, determining the magnitude of condensate holdup, and calibrating the condensate weigh system. (B.J.H.)

8650 CF-56-4-125
Oak Ridge National Lab., Tenn.
HRT VENT VALVE TEST. C. H. Gabbard and H. A. McLain. Apr. 17, 1956. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

In order to test the HRT vent valve, it was connected to a 80-kw steam generator, and 2000 psig saturated steam was allowed to pass through the vent valve at various openings. Results of the tests are tabulated, and experimental and calculated flow rates are shown graphically. (B.J.H.)

8651 CF-56-4-144
[Oak Ridge National Lab., Tenn.]
PROPOSAL TO RELOAD THE ORNL GRAPHITE REACTOR WITH ENRICHED URANIUM. Apr. 23, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8652 CF-56-4-167
Oak Ridge National Lab., Tenn.
PROCEDURE FOR INITIAL HYDROSTATIC TEST ON HRT FUEL AND BLANKET SYSTEMS. H. E. Williamson. Apr. 20, 1956. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

8653 CF-56-4-186
Oak Ridge National Lab., Tenn.
PROCEDURE FOR CLEANING THE LOW AND HIGH-PRESSURE FUEL AND BLANKET SYSTEMS WITH TRISODIUM PHOSPHATE HRT ENGINEERING TEST PROCEDURE. H. K. Search. April 27, 1956. 6p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

8654 CF-56-4-190
Oak Ridge National Lab., Tenn.
INTERNAL RECOMBINATION IN THE HRR. R. E. Aven. April 26, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An introductory study has been made to estimate the amount of $CuSO_4$ catalyst needed in the HRR to internally recombine the radiolytic gases produced. Design criteria for the study are a power of 500 Mw, core volume of 12,000 liters (10 ft core vessel), total volume of fuel loop of 36,000 liters, fuel concentration of 2 gms U-235/liter, and temperature rise across core of 50°C. Minimum copper re-

quirements for 100% internal recombination and for 80% internal recombination are given for three operating temperature and pressure ranges in the HRR. (auth)

8655 CF-56-4-214
Oak Ridge National Lab., Tenn.
OPERATING PROCEDURE FOR HEATING THE HRT HIGH-PRESSURE FUEL AND BLANKET CIRCULATING SYSTEMS USING STEAM FROM THE BUILDING SUPPLY AND PACKAGE BOILER. B. H. Hamling. Apr. 13, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

A recommended operating procedure for heating the high-pressure fuel and blanket circulating systems with steam from the building supply and the package boiler is given. An estimated minimum heating time of 4.4 hours and a minimum of 7900 lbs of steam is required for heating the HRT systems from 50° to 280°C. The heating rate is held to 100°F/hr for safety purposes. (M.H.R.)

8656 CF-56-5-40
Oak Ridge National Lab., Tenn.
DELAY TIME REQUIRED TO REDUCE D_2 EXPLOSION HAZARD DURING HRT DUMP. R. E. Aven and M. W. Rosenthal. May 1, 1956. 22p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

A study has been made to determine the time that a dump should be delayed after the HRT power shutdown has been initiated. A delay period is needed to allow time for the recombination of some of the radiolytic gases dissolved in the fuel solution. This delay time is, therefore, a function of copper concentration, rate of radiolytic gas production during the shutdown period, and permissible D_2 concentration in the gas phase of the HRT during a dump. The normal behavior of the system during the shutdown period has been estimated. (auth)

8657 CF-56-5-55
Oak Ridge National Lab., Tenn.
PRESSURIZER THERMAL INSULATION. M. C. Lawrence. May 1, 1956. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Thermal insulation needs for both the core and blanket pressurizer, based upon allowable heat losses determined from system needs and heating capability, are discussed. Calculations and graphs are included. (M.H.R.)

8658 CF-56-5-69
Oak Ridge National Lab., Tenn.
THE RELATIONSHIP OF SLURRY PHYSICAL PROPERTIES TO THE ENGINEERING CHARACTERISTICS OF A SLURRY SYSTEM. D. G. Thomas. May 11, 1956. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

In a reactor system the principal components affected by slurry properties are the blanket vessel, pressurizer, heat exchanger, and dump tank. The particular properties that affect the operation of these components are: caking, degree of flocculation, foaming, and slime formation. These properties are related to the characteristics of components in a reactor system through experience gained in the operation of slurry loops. It is pointed out that the optimum slurry for one component may not necessarily be the optimum for another. A program is proposed to determine quantitatively those slurry properties important for the specification of a slurry to meet the engineering requirements of a reactor system. (auth)

8659 CF-56-5-70
Oak Ridge National Lab., Tenn.
"VACUUM SWEEPER" TESTS FOR HRT-CP. R. W. Winget. May 8, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

It was recommended that a means be provided for preventing gross contamination of the shielding water by removing the water in the immediate vicinity of the newly opened flange requiring maintenance. Since a pump requires maintenance, it was thought that the 2-in. Penberthy jet already installed in the sump of cell C could be modified to provide suction for a portable vacuum system. The normal 2-in. suction line of the jet was changed to a 1-in. suction line and a second 1-in. suction line was added running parallel to the cell floor and mounted 90° from the normal suction line. With 80 psig steam to the jet, a 16.5 gpm flow rate was attained through a $8\frac{1}{2}$-in. diameter 1-in. stainless steel tubing suction ring with 24 equally spaced holes with a total cross sectional area of 1.58 inches^2. This flow was in addition to the flow through the 1-in. normal suction of the jet. By modifying the jet, no valves were necessary to operate either the portable suction side of the jet or the normal suction side of the jet since both will be used simultaneously. The minimum flow recommended for the "vacuum sweeper" technique was 10 gpm as compared to 16.5 gpm obtained in this system. Potassium permanganate dye used to simulate activity was pulled through the ring without any difficulty in a mockup of flange breaking in cell B of 7500. (auth)

8660 CF-56-5-89
Oak Ridge National Lab., Tenn.
CALCULATED INTERNAL SURFACE AREAS IN THE HRT HIGH-PRESSURE FUEL AND BLANKET SYSTEMS. R. Van Winkle. May 10, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Detailed listings of all the components of the systems and their areas are given. (M.H.R.)

8661 CP-308
[Chicago. Univ. Metallurgical Lab.]
EXAMPLES FOR PRESSURE DROP CALCULATIONS IN PARALLEL FLOW HELIUM COOLING. B. T. Feld and L. Szilard. June 18, 1942. Decl. Feb. 15, 1957. 4p. Contract [W-7401-eng-37]. (A-350). $1.80(ph OTS); $1.80(mf OTS).

Pressure drop calculations are shown for He-cooled power plants in the range 30 to 400 Mw. He pressures of 1 and 10 atm. are used. (D.E.B.)

8662 CP-2774
Chicago. Univ. Metallurgical Lab.
FINAL REPORT ON SWELL DETECTION BY PUSHER METHOD. W. Bradford Shank and M. Frankel. Nov. 11, 1944. Decl. Feb. 16, 1957. 41p. Contract [W-7401-eng-37]. $7.80(ph OTS); $3.30(mf OTS).

Experiments were performed to evaluate the feasibility of detecting swelled slug elements in Hanford reactors by measuring the force necessary to push the slug column in its tube. These experiments are described. The results indicate the method presents no unsolvable problems. (D.E.B.)

8663 CP-3493
[Chicago. Univ. Metallurgical Lab.]
TEMPERATURES AND THERMAL STRESSES IN PILE FUEL RODS. Clyde A. Hutchison, Jr. Apr. 15, 1946. Decl. Feb. 7, 1957. 25p. Contract W-7401-eng-37. $4.80 (ph OTS); $2.70(mf OTS).

The general equations relating temperatures and thermal stresses in pile fuel rods to their physical properties are given for the approximation of infinitely long rods. The relationships between the fuel rod stresses and those in a cooling cylinder are given. The available measurements of pertinent physical properties are summarized. Numerical calculations of temperatures and stresses are made for the case of a beryllium oxide, uranium oxide pile. (auth)

8664 DC-51-8-7
General Electric Co. Aircraft Nuclear Propulsion Dept., Cincinnati.
VOLUME OF THE NOBLE GASES FORMED IN FISSION. William H. Long. [1951]. Decl. Feb. 16, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

An approach to the problem of pressure build-up in a fuel element caused by the evolution of fission-product gases is presented. The decay equations for the number of nuclei formed as a function of time were solved and evaluated. The number of nuclei was then transformed to a STP volume of gas. The decay chains were simplified in that where the Te half life is short compared to the succeeding I half life, the Xe was taken as the direct daughter of the I. The Xe and Kr volume/g of U^{235} vs. the time are given for four power levels. The volume-vs.-time curves for individual Xe isotopes are given. A curve for total pressure vs. particle porosity for three power levels is presented. (G.Y.)

8665 DOW-NR-55001-1-S
Dow Chemical Co. Nuclear Research Lab., Midland, Mich.
PROPOSED REACTOR FOR POWER COMBINING FEATURES OF LMFR-SGR. Feb. 1, 1955. Decl. Feb. 25, 1957. 210p. Contract AT(11-1)-329. $0.40(OTS).

A program is outlined for the design and development of a commercial reactor to produce power and process steam. It is recommended that the basic structure follow the design of the Sodium Graphite Reactor but should be adapted for liquid metal fuel and equipped with a blanket. Ways and means are suggested for reducing developmental steps, simplifying installation, and cutting costs. If all liquid fuel experiments are complete failures, the basic structure can be converted into a sodium graphite reactor. (M.P.G.)

8666 HW-7-5660
[Hanford Works, Richland, Wash.]
APPARATUS FOR CHATTERING STUDIES. Final Report [on] Problem Assignment No. 3-P. C. P. Cabell. Jan. 17, 1947. Decl. Feb. 23, 1957. 6p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Suitable apparatus for studying chattering was built. It consists basically of a process tube with a pair of Lucite windows. Movement of slugs is observed through the windows, and simultaneous static pressure readings show the effect of slug movement on pressure drop through the tube. The apparatus was tested with various types of slugs and its usefulness has been demonstrated. (auth)

8667 HW-23207
[Hanford Works, Richland, Wash.]
WEIGHTING FACTORS FOR RADIALLY FLATTENED PILES. R. W. Woodruff. Jan. 10, 1952. Decl. Feb. 21, 1957. 4p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

In order to calculate pile reactivity changes which result from different pile loadings, values of $\int (\text{flux})^2 d(\text{volume})$ were found for a cylindrical pile having circular flat zones and effective radius of 598 cm. (B.J.H.)

8668 HW-25084
Hanford Works, Richland, Wash.
EXAMINATION OF DISCHARGED PROCESS TUBES. R. S. Dalrymple. July 21, 1952. Decl. Feb. 27, 1957. 7p. Contract [W-31-109-eng-52.] $1.80(ph OTS); $1.80(mf OTS).

Procedures for the examination of discharged process tubes are described briefly, and examples of tube wall corrosion are illustrated. (B.J.H.)

8669 HW-25614
Hanford Works, Richland, Wash.
CRITICALITY CONDITIONS FOR 1.75 PER CENT ENRICHED URANIUM SLUGS. John O. Erkman. Sept. 11, 1952. Decl. Mar. 12, 1957. 4p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

8670 HW-27551
[Hanford Atomic Products Operation, Richland, Wash.]
GENERAL FLATTENING AND POWER CURVES. Paul F. Gast. Mar. 30, 1953. Decl. Feb. 19, 1957. 9p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The optimum number of tubes for a given pile design may be ascertained from a determination of flattening and power characteristics. The methods of this determination from basic pile data are outlined. (D.E.B.)

8671 HW-30422
Hanford Atomic Products Operation, Richland, Wash.
VARIATION OF DIFFUSION LENGTH WITH SIGMA-PILE SIZE. E. Z. Block and D. E. Davenport. Jan. 4, 1954. Decl. Feb. 20, 1957. 20p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Four stacks of rough graphite were erected successively on a steel layup pad. They were each 9 feet high and were of the following nominal base dimensions: 4 ft × 4 ft; 8 ft × 8 ft; 12 ft × 12 ft; and 16 ft × 16 ft. The diffusion length was measured vertically in each and horizontally in the 12-ft square and 16-ft square stacks, using a Po-Be neutron source and a BF_3 proportional counter. The arithmetic average of the six traverses was 56.3 cm with a total spread of 1.8 cm. Since this spread conformed approximately to the overall uncertainty, it was concluded that the experiment did not detect any size-dependency of the diffusion length of a graphite medium when measured and calculated by equations of normal diffusion theory. (auth)

8672 HW-37208
Hanford Atomic Products Operation, Richland, Wash.
LIQUID MAGNESIUM AS A COOLANT FOR THERMAL REACTORS. J. R. Triplett. June 10, 1955. Decl. Feb. 23, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

A preliminary investigation was made of the possibilities of using liquid Mg as a coolant for thermal reactors. The reasons for considering Mg as a coolant are brought out by comparison of its relevant properties with those of liquid Na. It appears that Mg would offer some advantages above 1200°F. (B.J.H.)

8673 IDO-16019
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
EFFECT OF IMPURITIES IN COOLING WATER ON REACTIVITY OF THE MTR. G. H. Hanson. June 12, 1952. Decl. Mar. 14, 1957. 8p. $1.80(ph OTS); $1.80 (mf OTS).

The decrease in reactivity of MTR resulting from the presence of dissolved and suspended impurities in the reactor cooling water is $\sim -3 \times 10^{-5}$, which is negligible. Results are tabulated. (auth)

8674 IDO-16025
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
"BULK PARAMETERS" FOR MTR. George H. Hanson. Aug. 12, 1952. Decl. Feb. 23, 1957. 21p. $0.30(OTS).

Values of the Al/H_2O water ratio, fuel concentration and total absorption and fission cross sections for 0.025 ev and for thermal neutrons have been computed for 3 × 9, 4 × 9,

5 × 9, 5 × 7 and 5 × 5 lattice loadings. Fuel concentrations and total cross sections have also been computed for these lattice configurations using special fuel plates. The bulk constants for a 3 × 9 array of 23 fuel assemblies and 4 fuel shim rods fabricated with the present fuel plates are shown below. (auth)

8675 IDO-16046

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

NUCLEAR ASPECTS OF PROPOSED 19-PLATE FUEL ASSEMBLIES. G. H. Hanson and H. L. McMurry. Dec. 12, 1952. Decl. Feb. 26, 1957. 11p. [Contract AT(10-1)-205.] $0.25(OTS).

A new fuel assembly with nineteen 50-mil fuel plates and containing 168 gm of U^{235} for the MTR is discussed. The present "sweetened" fuel elements contain eighteen 60-mil plates and 168 gm of U^{235}. The thickness of the Al cladding for the fuel plates will be reduced from 20 mils to 15, and the thickness of the fuel-assembly side walls will be decreased from $^3/_{16}$ to $^1/_8$ inch. "Sweetened" shim rods with the present design will be used with the modified fuel assemblies. The mass of Al present in the fuel section of the new assembly is only 85% of that with the present design. The Al/H_2O volume ratio with a 3 × 9 loading of 19-plate assemblies is 0.581, whereas the ratio with 18-plate assemblies is 0.732. The total absorption cross sections of 3 × 9 loadings of 19-plate fuel elements and 18-plate elements are 9,180 and 9,126 cm^2, respectively, for 0.025 ev neutrons. The total fission cross section for each lattice is 6,137 cm^2. The calculated excess reactivity of a 3 × 9 loading of 19-plate fuel assemblies is about 2% greater than that with the present "sweetened" fuel elements or about 21%. The increased reactivity with the 19-plate fuel assemblies results chiefly from the 10% reduction in the "age" of the neutrons at thermal energy, i.e., reduction of the leakage of fast neutrons from the reactor. Since the "age" is relatively independent of the mass of U^{235} per fuel assembly, the substitution of 19-plate, 140-gm fuel elements for 18-plate, 140-gm units would also give an increase in reactivity of about 2%. (auth)

8676 IDO-16198

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

STATISTICAL ANALYSIS OF MTR FUEL ELEMENT NONDESTRUCTIVE ASSAY DATA. F. H. Tingey and F. P. Vance. Oct. 8, 1954. Decl. Feb. 23, 1957. 23p. Contract AT(10-1)-205. $0.30.

General-case equations for fuel element assay have been derived from the available data for several conditions: number of fuel elements in a group; single and duplicate measurements on each fuel element; number of measurements performed on the reference standard; and fuel core-length measurements. The over-all precision of the method can be improved by procurement of a fuel element assembled by taking one plate from each of the nineteen different alloy melts. (auth)

8677 IDO-16254

Phillips Petroleum Co., Atomic Energy Div., Idaho Falls, Idaho.

MTR TECHNICAL BRANCH QUARTERLY REPORT FOR THIRD QUARTER, 1955. J. R. Huffman. Dec. 30, 1955. Decl. Feb. 26, 1957. 20p. Contract No. AT(10-1)-205. $0.30(OTS).

Criticality and control tests on the Materials Testing Reactor are summarized. The possibility that a serious pressure rise would accompany the melting of a fuel element is investigated. Other boiling experiments are discussed. The γ intensity and heating values of spent fuel elements were investigated. The redesign of MTR shim rods is discussed. Ultrasonic welding may prove to be applicable to MTR fuel element fabrication. The fission neutron cross section for the reaction $Ni^{58}(n,p)Co^{58}$ was investigated, and it is concluded that there is considerable variation in the neutron spectrum with distance from the source and that the cross section for an unperturbed fission neutron spectrum is about 90 mb. The neutron absorption cross section of Re^{188} was also investigated. An analytical procedure for elements indicative of corrosion was developed. Critical mass calculations and void and temperature coefficients for SPERT were analyzed. Criticality problems in the chemical processing plant are briefly discussed. (For preceding period see IDO-16235.) (B.J.H.)

8678 K-1173

Massachusetts Inst. of Tech., Oak Ridge, Tenn. Engineering Practice School.

A PARAMETER STUDY OF THE K-49 REACTOR BLANKET. W. A. Stewart, J. R. Buchanan, Jr., R. E. Luedere, and A. B. Reynolds. Oct. 5, 1954. Decl. Feb. 23, 1957. 56p. Contract W-7405-eng-26, Subcontract 70. $0.50(OTS).

The ratio of Pu^{240} to Pu^{239} produced in the blanket of the proposed K-49 reactor has been derived as a function of reactor design and certain physical parameters. Two different reactor designs were selected, and a study is presented of the important parameters which affect the isotope ratio. The possibility of studying plutonium chemistry under reactor conditions, and determining plutonium solubility and precipitation rate using an in-pile loop in the LITR was also considered. (auth)

8679 KAPL-287(Rev.)

Knolls Atomic Power Lab., Schenectady, N. Y.

REACTOR MATERIALS, SECTION III [OF] PROGRESS REPORT NO. 41 [FOR] DECEMBER 1–31, 1949. Feb. 10, 1950. Decl. Mar. 6, 1957. 54p. Contract W-31-109-eng-52. $9.30(ph OTS); $3.60(mf OTS).

Solidification, solid-state transformation, recrystallization, and stress-strain anneal techniques for producing single U crystals were investigated. The latter method is considered favorable for producing large crystals. Theoretical and experimental considerations of U diffusion in Cu, Mo, and Zr are investigated. An apparatus was developed for measuring tensile creep at elevated temperatures under radiation conditions. Radiation-induced transformations in austenitic stainless steel were studied by a magnetic susceptibility method. The design of an apparatus for testing strain gauges at elevated temperatures is given. Data were obtained on Na attack of Pyrex from 125 to 500°C. Isothermal decomposition measurements were made for U–Cr alloys at 705°C. Stress-rupture and stress-fatigue data on 347 stainless steel were obtained in Na at 1000°F. (K.S.)

8680 KAPL-423

Knolls Atomic Power Lab., Schenectady, N. Y.

FAST FUEL ROD DESIGN AND ANALYSIS (FFR-2). C. F. Stahl, R. J. Fritz, and R. G. Kennison. June 5, 1956. Decl. Feb. 7, 1957. 51p. Contract W-31-109-Eng-52. $0.45(OTS).

This report gives a detailed analysis of the design variables in the core section of the fuel rod and then, by examination of these variables, reaches an optimum design. A thermal analysis of the design and of the breeder sections of the fuel rod and an analysis of the fluid flow through the reactor are included. Since the fluid flow depends on the condition of the entire reactor, the moderator rods are also analyzed. The breeder sections of the rod are designed with a maximum temperature that exceeds the allowable maximum. This excessively high temperature is, however,

localized in a small volume near the core, and by redesign it is shown that the temperature can be reduced to the design value. (auth)

8681 KAPL-M-WAN-7(Del.)
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT ON TRIP TO SPERT MEETING AT THE NATIONAL REACTOR TESTING STATION, IDAHO. W. A. Northrop. Sept. 23, 1955. Decl. Mar. 2, 1957. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

8682 LWS-24701
California Research and Development Co., Livermore, Calif.
FAST REACTOR COOLANT COMPARISON. H. A. Brammer. Dec. 15, 1952. Decl. Feb. 27, 1957. 11p. (CRD-R-17). $3.30(ph OTS); $2.40(mf OTS).

Calculations are made of the variation in critical mass of a fast reactor caused by several possible coolants. The coolants considered are Na, NaK, S, Pb, Bi, and Hg. Each system is calculated for U, Fe, Pb, and Bi reflectors. The derivation of the critical equation for a spherical core with an infinite reflector is included in an appendix. (B.J.H.)

8683 MonN-292
Clinton Labs., Oak Ridge, Tenn.
ANALYSIS OF THE MAXIMUM THERMAL STRESSES GENERATED IN THE FUEL TUBES DURING STEADY STATE OPERATION OF A HIGH TEMPERATURE PILE. W. L. Sibbitt and H. Etherington. Dec. 16, 1946. Decl. Feb. 28, 1957. 34p. $6.30(ph OTS); $3.00(mf OTS).

A theoretical investigation was made of the probable thermal stresses generated in cylindrical BeO and graphite fuel tubes during steady state operation in the Daniels Power Pile (Design I). Various tube dimensions were studied over a wide range of power levels, and resultant data are given. (B.J.H.)

8684 MonT-180
[Clinton Labs., Oak Ridge, Tenn.]
FACILITIES FOR THE CRITICAL SIZE EXPERIMENTS (HOMOGENEOUS AND HETEROGENEOUS). Final Report. W. M. Stanley, Jr. and R. E. Pahler. Jan. 17, 1947. Decl. Mar. 9, 1957. 53p. Project No. C-90-M. Contract W-35-058-eng-71. $10.80(ph OTS); $3.90(mf OTS).

At the instigation of the Physics Division the Technical Division was given the assignment of designing and coordinating the installation of facilities necessary for the determination of the critical size of a U^{235}–D_2O system. Critical size measurements, temperature coefficient experiments, and effects of control elements on pile reactivity will be determined with the facilities. Additional related data will be developed as are necessary for the final design of the proposed enriched pile. The facilities provided consist in general of the following: a vertical, cylindrical, stainless steel reactor provided with a tube sheet on top, from which the aluminum fuel tubes are suspended; a vertical, cylindrical, stainless steel reservoir and blow case, for pumping the D_2O moderator-reflector solution up to the reactor; a surge tank, for overflow of D_2O from the reactor; a safety and control rod system consisting of four safety rods and one control rod; and cold traps and driers, for providing dry compressed air and recovery of vented D_2O vapors. The reactor is located on the second floor of cell 7 of Building 205, which is thermally insulated and is controlled within ½°C in a range from 20°C to 50°C. The reservoir, cold traps and driers are located in cell 7, downstairs. The control room, cell six upstairs, contains the control panel, control rod positioning mechanism, a cathetometer for determining rod positions accurately, and storage facilities for fuel tubes. Briefly, the experiments will probably be run in the following manner. After the control and safety rods have been lowered into the reactor in the safety position, a predetermined number of fuel tubes will be mounted in the reactor tube sheet in the desired lattice pattern. The D_2O will then be pumped into the reactor to the desired level. With this completed, the system will be ready for a test. The control rod will be gradually raised and indications of the reaction checked at various control rod positions. An operating procedure complete with the necessary drawings is also incorporated in this report. (auth)

8685 NAA-SR-282
North American Aviation, Inc., Downey, Calif.
REACTOR EVALUATION QUARTERLY PROGRESS REPORT FOR MAY–JULY 1953. A. B. Martin, ed. Oct. 20, 1953. Decl. Mar. 2, 1957. 23p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported for the following reactor evaluation projects: economic-engineering studies of Na cooled, graphite moderated U^{233} power breeders; power costs for reactor sizes in the range around 1000 Mw; various start-up schemes for the reactor; calculations of changes in reactivity and in fuel composition of reactor systems fueled initially with mixtures of natural U, Pu^{239}, and Pu^{240}; reactivity change calculations for natural and low enrichment U systems formulated in terms of power series to facilitate numerical computation; a study of the fractional fuel burnups attainable in highly enriched reactors by the use of burnable poisons; and a preliminary study of the dependence of reactivity on burnup in fast reactors. (auth)

8686 NAA-SR-878
North American Aviation, Inc., Downey, Calif.
SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT FOR JUNE–AUGUST 1953. G. M. Inman, ed. Jan. 20, 1954. Decl. Mar. 4, 1957. 118p. Contract AT-11-1-GEN-8. $0.70(OTS).

Studies were completed on problems related to the nuclear performance of the sodium-graphite reactor such as temperature coefficients of reactivity, poison transients, neutron leakages, possible runaway during start-up, and the variation of conversion ratio with burnup. Some of these studies were specifically for the 30 megawatt SGR pilot plant. In connection with the reactor engineering work, an investigation was carried out in order to optimize the cooling system and power plant conditions in the full-scale plant. Also, the desirability of operating with two different fuel element diameters in a single reactor and of operating with more than one reactor as the heat source for a single power plant was evaluated. Design and experimental work was conducted in regard to problems in the use of stainless steel coolant tubes, an inner steel core tank to protect the graphite from the sodium coolant, and the alternative arrangement of Zr cans containing cell-sized stringers of graphite. Materials work was initiated or continued on the joining of Zr and stainless steel, the carburization of stainless steel, the weldability of stainless steel and Ni, mass transport by Na in stainless steel, and the properties of U fuel alloys. The fuel rod loading apparatus was successfully used to prepare a seven-rod fuel element of U slugs in stainless steel jackets and bonded by Na. As alternate power reactor design studies the use of S and diphenyl as coolants was investigated. Attention was also given to the possible use of some hydrocarbon liquid as the moderator. On the basis of present information, reactor arrangements making use of such materials do not appear capable of the high performance expected from the sodium–graphite combination. A major integrated effort was placed on the preliminary design and proposed program for the Sodium Reactor

Experiment (SRE). This is envisaged as a tank-type reactor with initial operation using graphite moderator canned in Zr and with Na as the coolant. The design stresses maximum experimental flexibility and has no provision for the generation of electricity. Its primary purpose is for development work on the technology of fuels, moderators, coolants, and general reactor components. While the SRE arrangement permits a variety of experiments to be performed relative to high-temperature, low-pressure reactor core concepts, the study of problems pertinent to the Na-cooled, graphite-moderated arrangement will be emphasized. (auth)

8687 NAA-SR-956
North American Aviation, Inc., Downey, Calif.
SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT [FOR] SEPTEMBER–NOVEMBER, 1953. G. M. Inman, ed. July 1, 1954. Decl. Mar. 2, 1957. 105p. Contract AT-11-1-GEN-8. $0.65(OTS).

Two sodium-graphite central station power plant designs were investigated. The first design is for an enriched U converter of 150 Mw, and the second is a Th thermal breeder of 300-Mw capacity. Engineering studies related to the general design of sodium-graphite reactors are reported. Metal, moderator, and coolant temperature coefficients of reactivity were evaluated for slow, fast, and very fast runaways in sodium-graphite reactors. The reactor is stable except at room temperature. Calculations for a startup accident in a 30-Mw SGR were carried out, the results of which emphasize the importance of full-power coolant flow rates during the startup process. By using the fact that the U β phase does not exist in a U–Zr alloy containing 20 at. % Zr, it is shown that substantial reduction in power cost per kwh may be achieved by operating the reactor at higher power. Neutron age was calculated for a number of hydrocarbons which show some promise for use as moderators. Tensile tests and welding properties of the projected Zr cans for the graphite moderator sections are reported. Strength tests were completed for flash-welded joints of Zr–347 stainless steel tubing. The joints are mechanically feasible for reactor use, and favorable corrosion results were obtained from static Na studies. Dynamic corrosion data are required. Weldability studies are reported for Ni, 304 L stainless steel, and a series of typical core tank joints. The design of a forced-convection loop for dynamic Na–Zr corrosion studies is described. A remote welding device was designed for the repair of coolant tubes. The dimensional stability of U and U alloys subjected to thermal cycling and quenching tests was studied in the α and high β regions. The unfavorable characteristics of the β phase transformation were noted, and it is suggested that under the reactor operating conditions, where the surface of the fuel slug is adequately cooled, a sufficient shell of α material might be maintained around an inner β phase to contain the dimensional instability. No significant differences were noted in the thermal and electrical conductivity of polycrystalline α-rolled U specimens which were oriented parallel and perpendicular to the rolling direction. Nondestructive testing of fuel element components at high temperatures is proceeding with an evaluation of techniques using eddy currents, temperature-sensitive paints, and infrared photography. A study of the production of thermal energy in an Fe thermal shield by capture processes has been extended to include the effect of fast-neutron leakage. The results are graphically presented for several shield thicknesses. The physical properties of several hydrocarbons were investigated in order to determine their suitability as biological shield coolants. Kerosene and toluene seem particularly promising. (K.S.)

8688 NAA-SR-1027
North American Aviation, Inc., Downey, Calif.
SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT [FOR] DECEMBER 1953–FEBRUARY 1954. G. M. Inman, ed. Aug. 1, 1954. Decl. Mar. 12, 1957. 71p. Contract AT-11-1-Gen-8. $0.55(OTS)

Engineering pertinent to the development of the sodium-cooled, graphite-moderated type of reactor was continued. This included work on problems related to the zirconium-canned moderator, low-enrichment uranium fuel, sodium piping, secondary coolant system, shielding, and the control and safety elements. A large fraction of the work was devoted specifically to problems of the proposed Sodium Reactor Experiment (SRE) configuration. In this connection, an integrated effort was initiated to prepare a complete preliminary design of the SRE by an early date. In addition, two alternate sodium-graphite reactor configurations were studied. One was an intermediate size, 145-thermal megawatt unit optimized for the production of low-cost plutonium. The second was a low power, 10-thermal megawatt unit intended for power production, but in which sodium circulation through the core was entirely dependent upon thermal convection. (auth)

8689 NAA-SR-1062
North American Aviation, Inc., Downey, Calif.
REACTOR SAFETY QUARTERLY PROGRESS REPORT [FOR] FEBRUARY–APRIL 1954. Oct. 1, 1954. Decl. Feb. 27, 1957. 29p. Contract AT-11-1-gen-8. $4.80 (pt OTS); $2.70(mf OTS).

The composition of the solder for the solder plug has been set as the Sn-Ag eutectic. Final tests on this solder show that life expectancies much longer than 6 mo are probable with the current design. The design of the heater tube to contain the solder plug has been settled. This consists of a Cu tube impregnated with $U^{235}O_2$. Arrangements have been made to have test specimens fabricated by powder metallurgy techniques. The equipment for the MTR in-pile test of trigger element response times has been largely completed and tested. The design of the complete inner capsule for the BF_3 safety element has been developed as well as the cladding technique. Mock-up elements have been tested in the Hanford test reactor to determine the control that may be obtained with elements of this type, although the analysis of the results has not been made. Prototype elements are also ready for testing in the test pile, except for loading with $B^{10}F_3$. Experiments have been designed and submitted for approval for production pile tests of prototypes. (auth)

8690 NAA-SR-1237
North American Aviation, Inc., Downey, Calif.
NEUTRON FLUX MEASUREMENTS IN THE MOCK-UP OF THE URANIUM PRODUCTION REACTOR. Richard Laubenstein, William Houghton, and David Martin. Sept. 1, 1955. Decl. Feb. 27, 1957. 79p. Contract AT-11-1-gen-8. $0.45(OTS).

Experimental studies have been made of the thermal and resonance neutron flux distributions in a core mock-up of the proposed Uranium Production Reactor. This reactor was designed for the production of U^{233} and contains U^{235} and Th loaded assemblies with light water as a moderator and coolant. The measurements were made using a flexible mock-up of a portion of the proposed reactor core structure which was surrounded by a buffer lattice of natural U. The core mock-up was installed on the thermal column of a water boiler reactor to provide a source of neutrons for the measurements. Experimental values of thermal utilization and resonance escape probability were obtained for two different variations of the proposed core structure and for

an alternate core design. The experimental results were compared with theoretical calculations. This comparison demonstrates the need for experimental measurements to obtain detailed thermal neutron flux distributions on complex assemblies. However, with the experimental results as a guide, calculational models and proper boundary conditions can be selected so that relative average thermal neutron fluxes over various regions of the assembly can be calculated with reasonable accuracy using one group diffusion theory. Experimental values of resonance escape probability tend to be lower than the theoretical values, although the agreement is reasonable when the relatively large uncertainty in the experimental measurements is considered. (auth)

8691 NAA-SR-1333
North American Aviation, Inc., Downey, Calif.
REACTOR PHYSICS QUARTERLY PROGRESS REPORT [FOR] NOVEMBER 1954–JANUARY 1955. Richard A. Laubenstein, ed. July 1, 1955. Decl. Feb. 7, 1957. 30p. Contract AT-11-1-GEN-8. $0.35(OTS).

Experimental measurements and theoretical analysis have been performed for a graphite thermal column which contains a regular array of air channels extending the length of the column. These channels have a marked effect on the relaxation length of thermal neutrons in the assembly. Further danger coefficient measurements have been made on U slugs of various enrichments in order to determine the utility of a H_2O boiler reactor for the danger coefficient testing of U. The H_2O boiler appears to be a sensitive instrument for use in determining variations in U enrichment. Results of calculations of the fast fission effect in hollow U slugs are summarized. The results also show the difference between a void and a moderating material in the central region. (auth)

8692 NAA-SR-1483
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
THE EXPERIMENTAL DETERMINATION OF THE RATE OF GAS EXPANSION IN A MARK II SAFETY ELEMENT. Denny L. Condotta. Mar. 15, 1956. Decl. Mar. 8, 1957. 54p. Contract AT-11-1-GEN-8. $0.50(OTS).

An equation was derived describing the adiabatic expansion of a perfect gas through an orifice under critical flow conditions. This equation was used to correlate experimental gas expansion data obtained employing a test apparatus similar to the Mark II Safety Element. Experimental data were obtained for three expansions with He and for three with BF_3. Although an instrumentation lag was indicated, the He expansions demonstrated the general applicability of the derived equation for expansions of an ideal gas. Correlations showed that if the pressure, temperature, and density data obtained from the BF_3 compressibility curve were used in the derived equation, the resulting calculated curve agreed very well with the experimental data. The error was usually on the conservative side; that is, the experimental time required to cause a specific expansion was less than the calculated time. The derived equation was used to calculate the expansion of BF_3 from the storage chamber of a Mark II Safety Element. The resulting curve indicated that the Safety Element would be 90% effective within 0.15 seconds after the solder plug had blown from the gas release tube. This figure is considered accurate within about 10% and represents an expansion starting at a temperature of 70°F. Higher temperatures would speed the gas expansion and thus would reduce the time required for the safety element to become effective. (auth)

8693 NAA-SR-Memo-38
North American Aviation, Inc., Downey, Calif.
HTR–BREEDING GAIN. C. Roderick. July 31, 1951. Decl. Feb. 28, 1957. 16p. Contract AT-11-1-gen-8. $3.30(ph OTS); $2.40(mf OTS).

In connection with HTR breeding gain, a study was made to determine the build-up of Th^{233}, Pa^{233}, and U^{233} when Th^{232} is irradiated with various thermal fluxes. The methods of calculation are given in detail. Breeding gains are shown as a function of irradiation time and nvt(flux-days). Parasitic capture and fractional amounts of Th^{233}, Pa^{233}, and U^{233} are both shown as a function of irradiation time. (B.J.H.)

8694 NAA-SR-Memo-58
North American Aviation, Inc., [Downey, Calif.].
NEUTRON FLUX DISTRIBUTION IN A LATTICE CELL. E. R. Cohen. July 16, 1951. Decl. Feb. 27, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

Curves plotted from calculations of thermal neutron flux in a square lattice of U rods in D_2O are given. (D.E.B.)

8695 NAA-SR-Memo-100
North American Aviation, Inc., Downey, Calif.
NEUTRON ABSORPTION AND RADIOACTIVITY OF COMMERCIAL SODIUM AND LEAD EXPOSED TO A NEUTRON FLUX. C. H. Robbins. Sept. 26, 1951. Decl. Feb. 27, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

In order to compare Pb and Na as possible reactor coolants, calculations were made of the neutron absorption and radioactivity of these elements when exposed to a neutron flux of 10^{13} neutrons/sec/cm^2. Calculation methods are described in detail, and resultant data are tabulated. Results indicate that impurities cause little additional neutron absorption. Most of the induced radioactivity in commercial Na is due to Na^{24}, and the induced radioactivity in high purity Pb is due largely to Pb^{209}, Cu^{64}, Cu^{66}, As^{76}, and Sn^{123}. Dose rates around pipes carrying these metals are estimated. (B.J.H.)

8696 NAA-SR-Memo-145
North American Aviation, Inc., Downey, Calif.
PROGRESS REPORT FOR OCTOBER 1951. Nov. 30, 1951. Decl. Feb. 27, 1957. 20p. Contract AT-11-1-gen-8. $3.30(ph OTS); $2.40(mf OTS).

The status of design studies for four reactors is given. The reactor types are: graphite-moderated, Na-cooled, heterogeneous; D_2O-moderated, D_2O-cooled, heterogeneous; a high-temperature, gas-cooled power reactor, and a low-power heterogeneous research reactor. MTA radiation damage and beam studies are also reported. (D.E.B.)

8697 NAA-SR-Memo-153
North American Aviation, Inc., Downey, Calif.
PRELIMINARY DESIGN STUDY OF D_2O MODERATED Pb–Bi COOLED NATURAL URANIUM POWER REACTOR. J. De Felice. Dec. 10, 1951. Decl. Feb. 27, 1957. 12p. Contract AT-11-1-gen-8. $3.30(ph OTS); $2.40(mf OTS).

A preliminary design study of a D_2O moderated, Pb–Bi cooled, natural U power reactor is discussed. The total heat production of the reactor is designed to be 500 Mw. Loading and unloading are intended to be done under full power, thereby cutting down the shutdown time of the reactor. The reactor is about twice minimum critical size; therefore, some depleted U may be used in the central region of the reactor. (auth)

8698 NAA-SR-Memo-191
North American Aviation, Inc., Downey, Calif.
FEASIBILITY OF SUBSTITUTING STAINLESS STEEL FOR ZIRCONIUM AS A CLADDING MATERIAL FOR THE H_2O MODERATED PuP. E. F. Weisner. Jan. 29, 1952. Decl. Feb. 27, 1957. 15p. Contract [AT-11-1-GEN-8]. $3.30 (ph OTS); $2.40(mf OTS).

It has been shown that the substitution of 0.010-inch steel for 0.020-inch zirconium is economically feasible,

even desirable. This is true even if the enrichment is not increased and the lower conversion ratio is accepted. However, it has also been shown that increasing the enrichment to maintain the same conversion ratio is even more desirable. Therefore, the design for the pressurized water PuP reactor should entail the use of 0.010-inch stainless steel, and accordingly such a design is presented. (auth)

8699 NAA-SR-Memo-194
North American Aviation, Inc., [Downey, Calif.]
CONTROL ANALYSIS OF THE PuP NO. 30C REACTOR. C. Roderick. Feb. 7, 1952. Decl. Feb. 27, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The control analysis shows that 16 control rods of 1.7-cm effective radius are sufficient to shut down the PuP #30C Reactor. The rods are arranged so that 12 lie in a ring of 40-cm radius, 3 in a ring of 20-cm radius, and one in the center. Physical dimensions of various types of control rods are shown as a function of the effective radius. (B.J.H.)

8700 NAA-SR-Memo-277
North American Aviation, Inc., [Downey, Calif.].
GEOMETRY OF THE CONVERTER REACTOR. R[alph] Balent. Mar. 28, 1952. Decl. Feb. 28, 1957. 18p. $3.30 (ph OTS); $2.40(mf OTS).

It has been suggested that slab geometry should be considered for the Converter Reactor. The specific advantages which could be obtained with such a geometry were not made clear; therefore, a comparison of least volume cylindrical reactors and slab reactors has been made. It is shown that the cylindrical reactors give the highest conversion ratio as a function of critical mass for a production rate of 10 kg U^{233} per month and also the lowest total hold-up of fuel. This is true regardless of the loading system used. In comparing systems consisting of all external conversion with systems combining internal and external conversion, it is shown that the latter gives the highest conversion ratio and lowest total hold-up of fuel. The above conclusions are based on three fundamental assumptions, namely, the blanket catching efficiency is the same for all systems (80 per cent was used), the coolant flows in one of the long dimensions of the slab, and the fuel and Th are loaded uniformly throughout the core; i.e., no flattening of the neutron flux is considered. A few other aspects of reactor geometry are discussed. (auth)

8701 NAA-SR-Memo-385
North American Aviation Inc., Downey, Calif.
PROPOSED FLUX DISTRIBUTION MEASUREMENTS FOR THE CONVERTER REACTOR. R. A. Laubenstein. July 10, 1952. Decl. Feb. 28, 1957. 17p. Contract [AT-11-1-GEN-8]. $4.80(ph OTS); $2.70(mf OTS).

Construction of a subcritical mock-up of a portion of the core of the converter reactor is proposed for the purpose of making intracell flux distribution measurements. Approximately 250 kg of Th and 3 kg of enriched U would be used to mock-up various proposed designs of the reactor core. The water boiler would be used as a source of neutrons to drive the mock-up. Detailed specifications for the thorium and U^{235} required are appended. (auth)

8702 NAA-SR-Memo-507
North American Aviation, Inc., [Downey, Calif.]
BASIS FOR CRITICALITY CALCULATIONS FOR SLIGHTLY ENRICHED URANIUM, PLATE-TYPE PLUTONIUM CONVERTER REACTOR. R[alph] Balent. Dec. 2, 1952. Decl. Feb. 27, 1957. 15p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

Analysis of η, ϵ, p, and f for the UPR is treated in detail. Corrections to f are made in order to account for side plates in the fuel assembly, and corrections to k are made for fission product poisons. (K.S.)

8703 NAA-SR-Memo-669
North American Aviation, Inc., Downey, Calif.
FLUX DISTRIBUTION—CONVERTER REACTOR. E. Cohen. Apr. 17, 1953. Decl. Feb. 27, 1957. 20p. Contract [AT-11-1-GEN-8]. $4.80(ph OTS); $2.70(mf OTS).

The results of a four-region, two-group calculation of the flux distribution in the thorium uranium converter reactor (UPR) are presented. The calculation procedure is based on a matrix formulation, and certain computational difficulties are encountered in following that scheme exactly. It is therefore a secondary purpose of the present memo to indicate where these difficulties arise and how they may be, at least in part, avoided. (auth)

8704 NAA-SR-Memo-672
North American Aviation, Inc., Downey, Calif.
SHIM ROD DESIGN REVISION FOR UPR. (Ref: NAA-SR-207, Fig. III-9). H. L. Sletten. Apr. 21, 1953. Decl. Feb. 28, 1957. 8p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

On the basis of preliminary design calculations in the sheath temperature in the fuel section of the shim rod is discussed. Reduction of the resistance in the shim rod to coolant flow to provide more cooling is reported. The pressure drop in the shim rod at a flow velocity of 29 ft/sec should be reduced by 10 psi from 60 to 50 psi in order to do this. Calculations are presented to demonstrate how this can be accomplished. (auth)

8705 ORNL-267
Oak Ridge National Lab., Tenn.
REACTOR TECHNOLOGY PROGRESS REPORT FOR QUARTER ENDING MAY 31, 1949. E. C. Miller, ed. July 19, 1949. Decl. Mar. 6, 1957. 112p. Contract W-7405-Eng-26. $18.30(ph OTS); $6.00(mf OTS).

The erection and assembly of the reactor mock-up have been completed and the system has been placed in operation. No major difficulties developed. Minor adjustments are now being made, and preparations for the hydraulic and other operating tests are nearing completion. The report includes references to the progress in reactor design, corrosion studies, metallurgical engineering development, control and removal of air-borne particles, liquid metal heat transfer and coolants, control and auxilliary equipment, and shielding. (D.G.B.)

8706 ORNL-1121(Rev.)
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 15, 1951. W. E. Thompson, comp. Jan. 9, 1952. Decl. Mar. 22, 1957. 193p. Contract W-7405-eng-26. $1.10(OTS).

Status of design and construction work on the HRE is given, and operation of the HRE mockup is described. Corrosion studies reported include passivation of stainless steel, corrosion by reactor solutions of Zr, Ti, various stainless and carbon steels, and effect of corrosion inhibitors. Effects of pile irradiation on corrosion of stainless steels and on gas evolution in reactor solutions are reported. Observations on H_2-O_2 and D_2-O_2 recombination are given. Densitometry of the reactor solution is discussed briefly, and optical spectra of some neutron-absorbing glasses are plotted. Solubilities of Y, Ba, Pu, Ce, and La sulfates in reactor solutions are reported, and fission-product build-up and β activities are tabulated. (T.R.H.)

8707 ORNL-1554
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY

PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1953. W. E. Thompson, comp. July 10, 1953. Decl. Mar 25, 1957. 148p. Contract W-7405-eng-26. $21.30(ph OTS); $6.90(mf OTS).

Progress of the Homogeneous Reactor Project is reported according to the following outline: Part I. Homogeneous Reactor Experiment studies of chloride corrosion of stainless steel piping; chloride removal by AgCl; control-rod and reflector effectiveness; gas-production rate effects on reactivity; reactor characteristics, steam generation, water decomposition, and general stability when operating at 1000 kw; a design for an off-gas pump to minimize radioactive stack wastes. Part II. Boiling Reactor studies of criticality conditions and Pu production rates as a function of fuel enrichment, moderator ratio, and vapor fraction for UO_3-D_2O slurries and $UO_2SO_4-D_2O$ solutions; bubble nucleation by fission fragments; power removal in mock-up, electrically heated, volume-boiling systems. Part III. Intermediate Scale Homogeneous Reactor design studies of Cu^{+3} and Pt–alumina catalysis of H-O recombination; cooling-tower reactors; slug flow tests in a model core; pipeline gas separators; pulsafeeder and stellite piston pump tests; UO_2SO_4 and UO_2F_2 corrosion rates of various stainless steel alloys as a function of acidity, temperature, and velocity; stability of ThO_2 pellets in UO_2SO_4; corrosion testing by a torus rotator and by in-pile loops using Cu catalysts; static corrosion tests of synthetic gems and spinels in UO_2SO_4 solutions; susceptibility of 347 stainless steel and Ti to crevice and stress corrosion; welding techniques for 347 stainless steel; the physical and metallurgical properties of Ti and Zr–In alloys; H–O recombination mechanism in UO_2SO_4 solutions; the solubility of UO_3 in aqueous UO_2SO_4 at 175°C; phase studies on the hydrolytic or solubility precipitation of U and Cu in $CuSO_4$–, UO_2SO_4–, $UO_2SO_4-CuSO_4$–, $UO_2SO_4-CuSO_4-H_2SO_4$–, and $UO_2(Cr_2O_7)$–water systems; the solubility of H in UO_2SO_4 at high temperatures; the design of apparatus for solubility determination above 100°C; an apparatus for phase studies and corrosion measurements; fission product distributions; additional aqueous UO_2SO_4 vapor pressure data; production of UO_3 by the decomposition of $(NH_4)_4UO_2(CO_3)_3$; the crystal structure of $UO_3 \cdot H_2O$; the preparation and hydrolytic decomposition of UO_2CO_3; neutron irradiation effects on UO_3 slurries; ThF_4 and $Th(OH)_4$ as possible slurry materials; solid formation in $UO_3 \cdot H_2O$–water slurries above 200°C and elimination of this condition; test of UO_2CO_3 and $UO_3 \cdot H_2O-Na_2HPO_4$ slurries; comparison of fuel chemical processing procedures; stability in H_2O and 0.02M UO_2SO_4 at 250°C of ThO_2 pellets fired at 1700°C; dissolution of ThO_2 pellets. (K.S.)

8708 ORNL-1678

Oak Ridge National Lab., Tenn.

HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1954. W. E. Thompson, comp. Apr. 15, 1954. Decl. Mar. 7, 1957. 116p. Contract W-7405-eng-26. $0.70(OTS).

Homogeneous Reactor Experiment. An experiment with dissolved Cu as a recombiner for the decomposition gases, H_2 and O_2, proved successful. Further studies were made of the core-concentration effect on stable reactor operation. The generalized corrosion rate of the reactor stainless steel appeared higher but is thought to be due to operation in a corrosion-sensitive temperature region. The only operational difficulty experienced was the failure of a high-pressure, remote, liquid-level indicating device. Boiling Reactor Research. Additional results are reported from slip-velocity and natural-convection studies, and an investigation of separator pumps was initiated. General Reactor Studies. The design of a homogeneous test reactor (HRT) to replace the HRE, is presented. Results from a study on large aqueous U- and Pu-fueled homogeneous reactors for producing central station power are summarized. An alternate model to the straight-through core was tested. A full-scale model of the gas separator was tested. An electrolytic cell which was designed to produce stoichiometric gas for the catalytic recombiner test loop is described. Abrasion tests were made of thoria pellets against each other and against samples of brass, stainless steel, and Zr by fluidizing with H_2O. Pump construction and testing is summarized. Critical conditions for the proposed HRT have been calculated for several moderator mixtures in the core and for three blankets; heavy water, ThO_2 slurries, and low-enrichment UO_2SO_4 solutions. Control and instrumentation progress is summarized: Extensive data are presented from dynamic-corrosion tests of UO_2SO_4 solutions on stainless steel; Zr, Zr alloys, and Ti. An in-pile loop test to determine the effect of γ radiation on Graphitor specimens in UO_2SO_4 is discussed. In-pile loop development is summarized. Static-corrosion test results of Stellite, Inconel, Ta, Zircaloy II, and stainless and C steels in decontaminating solutions are presented. Dynamic-corrosion data of welded and unwelded austenitic stainless steel in UO_2SO_4 are given. Further impact testing of Zircaloy II after exposure in in-pile loops is reported. Preparation of Th oxide slurries from the oxalate, formate, and oxycarbonate is discussed. Abrasive and corrosive effects from circulating ThO_2 and Th hydroxide slurries in stainless steel loops are discussed; testing of materials other than stainless steel for the slurries is reported. Additional studies are reported on the CaF_2 process and alternate methods for removing rare earths from UO_2SO_4 solutions. (L.M.T.)

8709 TID-5392

Du Pont de Nemours (E. I.) & Co. Atomic Energy Div., Wilmington, Del.

NOISE LEVEL OF EXTERNAL HYDRAULIC SYSTEMS. NYX Test No. 1.2.5. Bruce Longtin. Apr. 1953. Decl. Mar. 7, 1957. 21p. Contract AT(07-2)-1. (DPXN-208). $4.80(ph OTS); $2.70(mf OTS).

8710 TID-10100

Pacific Northwest Power Group, Richland, Wash.

FIRST ANNUAL REPORT TO THE ATOMIC ENERGY COMMISSION. Apr. 1955. Decl. Mar. 5, 1957. 51p. $0.45(OTS).

A survey and study of reactor development activities have been undertaken to determine the feasibility of a commercial nuclear power plant. A water-cooled graphite-moderated reactor or WGR proposed by General Electric Co. was the first reactor type studied. An engineering analysis of the reactor is presented, including cost estimates, research and development problems, general description of plant, and operation and maintenance. It was estimated that the total cost of power would be over 6 mills per kwh. It was concluded that a water-cooled graphite-moderated reactor was not economically feasible. A program of future studies includes review of improvements and redesigns of the WGR and analyses of other types of power reactors including the Aqueous Homogeneous Reactor. (M.P.G.)

8711 WAPD-CP-1200

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].

CONTAMINATION AND DECONTAMINATION EFFECTS IN PWR. I. THE EFFECT OF SURFACE FINISH ON CONTAMINATION. L. A. Waldman. [July 1955]. Decl. Mar. 25, 1957. 29p. Contract AT-11-1-GEN-14. $6.30(ph OTS); $3.00(mf OTS).

Data presented indicates that surface finish is not an important variable with regard to contamination in the PWR primary coolant piping components. (auth)

8712 WAPD-CPM-1
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
RADIOCHEMISTRY OF FIRST PWR FUEL MATERIAL TEST—X-1 LOOP NRX REACTOR. Paul W. Frank, Kenneth H. Vogel, and Paul Cohen. Oct. 20, 1955. Decl. Mar. 4, 1957. 59p. Contract AT-11-1-GEN-14. $0.50 (OTS).

The First Defect Test in the (X-1) Loop of the NRX Reactor, AECL was initiated to determine: the physical effects of irradiation on defected clad specimens of fuel samples considered for a Pressurized Water Reactor (PWR), the extent of release of fission product activity to the primary coolant from a defected sample, and the distribution of fission product activity in a prototype PWR loop. The activity in the loop water rose rapidly at the start of the test. The steady-state value increased gradually throughout the test. The activity level remained within operational limits until an "activity burst" occurred. The high activity level in the loop water resulted in the decision to terminate the test. Subsequent examination of the fuel samples showed that the cladding on one of the defected UO_2 samples had swelled and split and the UO_2 had fractured. Analytical results indicated that the most important mechanism by which the loop was contaminated was by the deposition of particulate UO_2 with non-leachable fission products suspended throughout the UO_2 matrix. The largest decontamination factor was achieved during the first cycle of the decontamination run in which cold water was circulated through the loop. (auth)

8713 WAPD-CPM-2
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
RADIOCHEMISTRY OF SECOND PWR FUEL MATERIAL TEST—X-1 LOOP NRX REACTOR. Paul W. Frank, Kenneth H. Vogel, and Paul Cohen. [1956]. Decl. Mar. 5, 1957. 38p. Contract AT-11-1-GEN-14. $0.40(OTS).

Radiochemical analyses of loop water samples and of the ion-exchange resin on stream during the test indicated that those isotopes investigated could be divided into three groups in order of their relative contribution to the specific activity of the loop water: (1) I^{131}, Cs^{137}, Kr^{88}, Xe^{133}; (2) Sr^{89}, Sr^{90}, Ba^{140}; and (3) Zr^{95}, Ce^{141}, Ce^{144}, Y^{91}. The relative contribution of various isotopes to loop contamination was not determined. (auth)

8714 WAPD-FE-830
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
SUPPLEMENT TO PROPOSAL FOR CENTRAL THIMBLE TEST, WAPD-NRX-CR-5. J. H. Bach. May 24, 1955. Decl. Feb. 11, 1957. 11p. $1.80(ph OTS); $1.80(mf OTS).

The decision to use UO_2 in the PWR blanket and the resultant termination of the dual fuel element program resulted in modifications in the Chalk River Central Thimble Test proposed in WAPD-FE-704. Two UO_2 bundles to be tested represent the two types of assembly processes considered for the core. One of these is a welded tube sheet assembly and the other a mechanically joined assembly. Drawings and prints referring to the test are included. (C.H.)

8715 WAPD-MRP-42
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD NOVEMBER 1, 1953 TO JANUARY 1, 1954. Decl. Feb. 7, 1957. 41p. Contract AT-11-1-Gen-14. $0.40(OTS).

Design and control studies for the PWR are summarized and plate, rod, and pellet-type fuel elements are discussed. Corrosion and metallurgical studies are reported. (F.S.)

8716 WAPD-MRP-43
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD JANUARY 1, 1954 TO MARCH 1, 1954. Decl. Feb. 7, 1957. 48p. Contract AT-11-1-GEN-14. $0.45(OTS).

PWR fuel element design studies indicate the feasibility of U–Mo fuel alloy in a rod configuration. Burnout and mass transfer tests have been completed. U–Mo alloy preparation techniques are described, together with the fabrication of reactor plate, rod, and pellet fuel assemblies. Dynamic corrosion testing of PWR core materials and the characteristics of several corrosion inhibitors are reported, including U-base fuel alloys. Data are presented on the corrosion rate of unclad U–Mo alloy in 575°F H_2O. Accelerator irradiation experiments were performed on Zr in contact with H_2O flowing at 8.5 fps by 2-Mev electrons. UNIVAC calculations of ϵ and p for the PWR are reported. Further studies have been made on reactor control by Li_3BO_3 and the compatibility of this compound with other reactor materials. (K.S.)

8717 WAPD-MRP-45
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD APRIL 22, 1954 TO JUNE 3, 1954. Decl. Feb. 7, 1957. 60p. Contract AT-11-1-gen-14. $0.50(OTS).

Various techniques for the extrusion of Zr cladding components are discussed. Melting and fabrication of U–Zr alloys and melting of U–Mo alloys are treated in connection with the fabrication of Zr-clad U–Mo fuel elements. The cladding of fuel pellets by Zr indicates that resistance welding of hemispherical cups about the pellet will provide a low-cost fabrication process. The results of dynamic corrosion testing of PWR materials are reported, together with autoclave testing of $Li(OH)_2$ and Morpholine corrosion inhibition of carbon and low-alloy steels. Static corrosion tests were conducted on U–Mo, U–Mo–Nb, U–Mo–Pt, and several Zr alloys. Phase studies of the U–Mo system were begun. A procedure has been developed for the fabrication of U–Si compacts by powder metallurgical techniques. The hardness of Zircaloy-2 was investigated. The deposited activities of fission products from the corrosion of fuel elements were evaluated. Criticality calculations were made for shutdown rod positions, requiring the local flux peak at the edge of the water holes to be no greater than the flux at the core center with rods removed. Two-region critical experiments with the 3:1 H_2O–U lattice with 1.3% enrichment gave negative temperature coefficient of reactivity above 29°C. Chemical control development studies show that pumping requirements of the chemical control system in excess of 200 gpm has very little effect on the ability of the system to contain transients. Some corrosion studies of the control solutions (H_2O, boric acid, LiOH) on important reactor materials are included. For the given lattice system, the poison concentration must be measured to an accuracy of ±8.5% for a determination of k to ±0.1%. (K.S.)

8718 WAPD-MRP-46
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD JUNE 3, 1954 TO JULY 15, 1954. Decl. Feb. 7, 1957. 70p. Contract AT-11-1-GEN-14. $0.50(OTS).

Primary coolant pump development includes mainly detailed improvements in design and manufacture. Develop-

mental changes were made to valves for the Mark II primary coolant system. Preliminary design were completed and tests initiated on small strainers. Improved instrument prototype designs were completed for differential transmitters and high speed resistance thermometers. Preliminary tests are in progress on the properties and behavior of Monel as an alternate material for heat exchangers. Data on chloride stress corrosion tests on austenitic stainless steel are reported. New approaches in the manufacture of bimetallic joints show some promise. (L.M.T.)

8719 WAPD-MRP-50

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD NOVEMBER 18, 1954 TO DECEMBER 30, 1954. Decl. Feb. 7, 1957. 92p. Contract AT-11-1-GEN-14. $0.60(OTS).

Temperature, power, and reactivity transients for the PWR were calculated for a step change in steam demand from 100 to 25%. The properties of Mallinckrodt UO_2 powders were determined by sieve analyses and flow rate tests. A cold-pressed cylindrical UO_2 slug of 75% theoretical density was unaffected by 100-day exposure in 650°F water. Additional burnout data are given for a rectangular Ni channel. A study of the preparation and properties of bulk UO_2 as a PWR fuel material was begun. Tensile tests have been conducted on γ-quenched U−12 w/o Mo alloy to determine the effect of strain rate over a temperature range from room temperature to 700°F. The effect of aging at temperatures below 400°C on the corrosion of U−Mo and U−Mo−Pt alloys by 650°F water was investigated. Other corrosion measurements are proceeding with U−Mo−Nb and U−Mo−X alloys, quenched U−Zr−Nb, and Zircaloy. Decontamination procedures were tested for a group of U−Mo fuel specimens. A preliminary experiment on the decontamination of PWR liquid wastes, using clay and synthetic green sand was completed. Radiation effects on the dimensional stability of U−Zr alloy and the physical properties of U−Mo alloy were determined. Design studies for an annular seed and blanket PWR design were completed. A fuel loading for 3000 hr of full power operation is provided, with equilibrium Xe override at the end of life. Measurements of the temperature coefficient were made for a 3:1 lattice of 1.15% enrichment, using 0.6-in. diameter U−Mo rods. The thickness of the secondary concrete shield for the PWR was computed to be 145 cm. (K.S.)

8720 WAPD-MRP-53

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PRESSURIZED WATER REACTOR PROGRAM TECHNICAL PROGRESS REPORT FOR THE PERIOD MARCH 24, 1955 TO MAY 5, 1955. Decl. Feb. 7, 1957. 160p. Contract AT-11-1-GEN-14. $0.85(OTS).

The current PWR plant and core parameters are listed. Primary coolant volume surges were computed using values of -1 and -2×10^{-4} Sk/°F as the reactor temperature coefficient. Analytical studies have been completed of possible pressure and temperature build-up in the vapor container resulting from a primary coolant piping rupture. The rate of pressure rise in water-logged UO_2 rods during an increase of power level has been studied. Some results are presented from a study of core hot channel conditions after loss of coolant flow. It was found that the process of agglomeration of UO_2 powder using polyvinyl alcohol as binder is sensitive to variations in mass and in manipulative techniques. Extrusion development, corrosion testing, and end closure studies, of U−12 wt. % Mo fuel rods are reported. Corrosion behavior of UO_2 compacts sintered in H and in A is summarized. Measurements of the surface area and density of UO_2 powders prepared by different methods have been made to evaluate the validity of the measuring techniques. Work on the reaction of Zr with UO_2 has continued. The effect of additions of Ni and Pt to water on the corrosion of U−12 wt. % Mo alloy is reported. Additional data on the water corrosion of U−Mo− Pt and U−Nb alloys are summarized. The effects of low temperature annealing, annealing time, cladding, homogenization, melting practice, extruding and drawing, fabrication conditions, and irradiation on the corrosion of U−Mo alloys are reported. Phase studies are being made on U−Mo alloys. Fission product removal studies have included analysis of the distribution of fission products from irradiated U−Mo fuel elements corroded in a static autoclave and a study of the adsorption of fission product activity on surface finishes. Irradiation tests are described on clad and unclad U−Mo alloys, clad and unclad U−Nb alloys, unclad U−Si alloys, and Zr-clad UO_2. Results of experimental measurements of the escape of fission product gases from irradiated UO_2 specimens are reported. Analysis of the reference PWR seed and blanket design was continued. The temperature coefficient of reactivity for a U−Mo blanket at the end of life with equilibrium Xe was calculated to be -4×10^{-4}/°F at 525°F. A 3 group one-dimensional calculation for the flux distribution and Pu formation in the blanket has indicated that the expected high rate of Pu formation in the blanket adjacent to the seed will be realized. Preliminary values of the buckling, migration area, and temperature coefficient on the first critical experiment on the PWR flexible assembly were obtained. (M.P.G.)

8721 WAPD-PWR-CP-1832

Westinghouse Electric Corp. Bettis Plant, Pittsburgh.

PROPOSAL FOR DYNAMIC IN-PILE CYCLING TEST OF PWR FUEL MATERIALS AT THE MTR — WAPD-25. W. Schwartz. Feb. 15, 1956. Decl. Mar. 5, 1957. 20p. $0.30(OTS).

Refer also to abstracts 8469, 8500, and 8739.

NUCLEAR TRANSFORMATION

8722 MTA-6

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

FISSION AND CAPTURE EVENTS IN 12 INCH X 12 INCH CROSS SECTION URANIUM AND THORIUM TARGETS BOMBARDED WITH 190 MEV DEUTERONS AND 340 MEV PROTONS. H. G. Hicks. Jan. 16, 1953. Decl. Feb. 27, 1957. 36p. Contract AT(11-1)-74. $6.30(ph OTS); $3.00 (mf OTS).

Appropriate foils were placed in 12-in square blocks of U and Th metal of varying lengths, the blocks bombarded, and the foils analyzed radiochemically for fission and capture events. Emergent neutrons were measured by capturing them in a large tank containing $MnSO_4$ solution. Multiplication of neutrons by fast secondary neutrons was confirmed. When moderator was present in the blocks, multiplication was decreased. When bombarding with 190-Mev deuterons, the total yield of neutrons reached a maximum when the target was 10 in. deep. When Be instead of U or Th was used to produce neutrons from the charged particle beams, the total neutron yield tended to be somewhat lower. Bombardments using 340-Mev protons showed that the neutron yield as well as the number of fission events increased with bombarding energy. Distribution of fission events in the targets indicated expected heat loads. Two types of secondary fission processes were shown to occur, an asymmetric, low energy type induced by neutrons with energies below 10 Mev, and a symmetric, high energy type induced by neutrons with energies above 20 Mev. (auth)

8723 UCRL-951
California. Univ., Berkeley. Radiation Lab.
NEUTRON YIELD IN URANIUM VS. ENERGY OF DEUTERONS AND PROTONS. H. Brown. Oct. 11, 1950. Decl. Mar. 5, 1957. 6p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

The semi-empirical curve of neutron yield vs. deuteron energy is recalculated for use in predicting yields for deuterons on U in the extrapolated energy region. (T.R.H.)

8724 UCRL-1676
California. Univ. Berkeley, Radiation Lab.
ENERGY SPECTRUM OF NEUTRONS FROM 3–10 MEV. Frank Adelman. Feb. 11, 1952. Decl. Mar. 4, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

The energy distributions of neutrons in the 3- to 10-Mev energy range resulting from the deuteron and proton bombardment of thin U targets were investigated by the measurement of proton recoils in photographic plates. Plotted data represent the spectrum observed at 135° from the 190-Mev deuteron beam. For comparison the fission spectrum is also plotted. Differences are discussed briefly. (C.H.)

8725 UCRL-2319
California. Univ., Berkeley. Radiation Lab.
THE PRODUCTION OF 320 MEV DEUTERONS BY HE^3 STRIPPING. John Ise, Jr., Robert V. Pyle, Donald A. Hicks, and Robert M. Main. Aug. 31, 1953. Decl. Mar. 6, 1957. 19p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Deuterons can be accelerated in the 184-in. cyclotron to 190 Mev. To produce higher energy deuterons, doubly charged He^3 ions are accelerated to 510 Mev, and a fraction of them are stripped in an internal target, yielding deuterons with an average energy of 340 Mev. A current of 5×10^{-13} amp is obtained in the experimental "cave" external to the cyclotron shielding. Range measurements indicate that these deuterons have an average energy of about 320 Mev. The high cost of He^3 necessitates an efficient gas recovery system. The exhaust from the cyclotron diffusion pumps is passed through activated charcoal traps which are cooled to liquid-N temperatures, and the gas which is not adsorbed (presumably He) is returned to the cyclotron source. The loss from all causes is about 0.8 cc/hr, which represents a cost of $30/hr of running time. (auth)

8726 UCRL-2705
California. Univ., Berkeley. Radiation Lab.
SUMMARY OF RECENT EXTERNAL NEUTRON-YIELD MEASUREMENTS BY THE $MnSO_4$ TANK METHOD. Walter E. Crandall and George P. Millburn. Sept. 17, 1954. Decl. Mar. 7, 1957. 11p. Contract W-7405-eng-48. $1.80(ph OTS); $1.80(mf OTS).

Neutron yields from a number of targets bombarded by deuterons (320 and 190 Mev), neutrons (160 and 90 Mev), He^3 (490 Mev), and protons (340 Mev), were measured for different target geometries by the $MnSO_4$ tank method. (W.L.H.)

PARTICLE ACCELERATORS

8727 CF-56-3-112
Oak Ridge National Lab., Tenn.
NOTES ON AN AZIMUTHALLY VARYING FIELD (AVF) CYCLOTRON. F. T. Howard, ed. Mar. 28, 1956. 38p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Excerpts from ORNL progress reports and abstracts of technical papers on the development of an azimuthally varying field cyclotron are collected. Discussions are included on such topics as accelerator studies at 1 Bev, a 1-Bev proton cyclotron, magnetic phase compensation in cyclotrons and a fixed-frequency electron cyclotron with azimuthally varying magnetic fields. (B.J.H.)

8728 CRD-T4A-85
California Research and Development Co., Livermore, Calif.
EFFECT OF LATTICE THICKNESS AND REFLECTOR MATERIALS ON THE PRODUCTION OF A WATER MODERATED THORIUM FUELED MULTIPLYING LATTICE. W. H. Harker. Feb. 6, 1952. Decl. Feb. 14, 1957. 11p. (LWS-24187). $3.30(ph OTS); $2.40(mf OTS).

A series of two-group two-medium calculations were carried out to determine the best reflector material for use with H_2O-moderated lattices. Lattice thicknesses of 15, 20, 25, and 30 cm backed by D_2O, H_2O, BeO, Fe, and graphite were investigated. (D.E.B.)

8729 LRL-69
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
NEUTRON PRODUCTION FROM HIGH-ENERGY CHARGED PARTICLE BOMBARDMENT OF INFINITE URANIUM TARGETS. C. C. Old. Dec. 1953. Decl. Feb. 28, 1957. 26p. Contract AT(11-1)-74. $4.80(ph OTS); $2.70(mf OTS).

The mechanisms of neutron production in large U targets when bombarded by high-energy protons and deuterons are reviewed and discussed. A neutron-yield kernel is developed, and total neutron yields are calculated at several bombarding energies for deuterons and for protons and compared in two instances with experimental data obtained with the UCRL 184-in. cyclotron. (auth)

8730 LRL-119
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
MTA PROJECT ACCELERATOR QUARTERLY PROGRESS REPORT FOR JULY THROUGH SEPTEMBER 1953. May 1954. Decl. Feb. 28, 1957. 30p. Contract AT(11-1)-74. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported in the development of a high-current linear accelerator. After correcting the voltage holding problems it was possible to accelerate proton beams with peak currents of 200 ma in short pulses and at pulse rates up to 8.2/sec, or continuous wave beams up to 50 ma for short periods. Minor design improvements are described. Mark I operation and performance for the quarter are reviewed. Preliminary studies are reported on an accelerator which would operate at a frequency of about 50 Mc. MTA component development is described. (C.H.)

8731 LWS-22248
California Research and Development Co., Livermore, Calif.
MEASUREMENT OF AXIAL FIELD OF MARK I BY THE "GLOWING BALL" METHOD. S. P. Stone. July 9, 1952. Decl. Feb. 28, 1957. 8p. $3.30(ph OTS); $2.40(mf OTS).

8732 LWS-24418
California Research and Development Co., Livermore, Calif.
HEAT GENERATION AND THERMAL STRESSES IN VACUUM VESSEL. R. C. Gerber and F. J. Duvall. May 28, 1952. Decl. Feb. 26, 1957. 14p. (CRD-T1-116). $3.30(ph OTS); $2.40(mf OTS).

The calculations indicate that additional cooling may be required for portions of the vacuum vessel with large stiffener sections located close to the primary and secondary targets. Better data are required for final design covering the gamm ray irradiation intensity from the primary and secondary targets, including spatial and

energy distributions. In addition, the calculations must be repeated for arrangements differing from the geometry assumed in this report. (auth)

8733 UCRL-1384
California. Univ., Berkeley. Radiation Lab.
CALCULATION OF SHIELDING FOR MARK II ACCELERATOR. B. J. Moyer. June 30, 1951. Decl. Mar. 5, 1957. 10p. Contract [W-7405-eng-48]. $3.30(ph OTS); $2.40 (mf OTS).

Numerical integration methods were used to calculate the shielding for the Mark II accelerator, and resultant data are given in graphical form. Data are also included on deuteron losses in drift tubes and magnets and neutron attenuation in concrete. (B.J.H.)

8734 UCRL-2021
California. Univ., Berkeley. Radiation Lab.
MTA, A-12 DEVELOPMENT, DRIFT TUBES—RF AND BEAM GEOMETRY. COMPARISON OF A-12 TO A-24. A. Schelberg and S. Kitchen. Nov. 17, 1952. Decl. Mar. 7, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

8735 UCRL-2318
California. Univ., Berkeley. Radiation Lab.
MTA QUARTERLY PROGRESS REPORT [FOR] MARCH, APRIL, AND MAY 1953. Aug. 28, 1953. Decl. Mar. 4, 1957. 19p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Topics relating to target and accelerator research are discussed as follows: inelastic deuteron cross sections for Al, Be, Ca, Cu, Pb, and U; external neutron yields from U bombarded by 300-Mev deuterons; design of a liquid scintillation neutron detector; neutron evaporation from excited nuclei; beam dynamics and preaccelerator design for the Mark I MTA. (J.E.D.)

Refer also to abstract 8438.

RADIATION ABSORPTION AND SCATTERING

8736 CF-56-1-9
Oak Ridge National Lab., Tenn.
MULTIPLE GAMMA-RAY SCATTERING IN WATER. T. A. Welton. Jan. 4, 1956. 16p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Calculations of the multiple γ-ray scattering in water are given. Applications of the calculations to reactor shield designs are briefly indicated. (B.J.H.)

RADIATION EFFECTS

8737 CC-1677
[Chicago. Univ. Metallurgical Lab.]
THE EFFECT OF RADIATION ON THE CORROSION OF METALS BY WATER. (Problem No. 322 MLC 2213). A. O. Allen, M. C. Bowman, N. Goldowski, R. G. Larson, and L. Treiman. July 6, 1944. Decl. Feb. 16, 1957. 39p. Contract [W-7401-eng-37]. (A-2413). $6.30(ph OTS); $3.00(mf OTS).

Long-time tests have been made on the effect of various types of radiation on the corrosion of 2S aluminum in simulated pile water. In no case was any acceleration of corrosion by the radiation observed; the effect of radiation, if any, appeared to be a protective one. Deuteron irradiation did accelerate the corrosion of mild steel at low flow rates in hot water of pH 6 to 7, but no appreciable effect was observed with copper, stainless steel, or tuballoy. The general theory of the effect of radiation on corrosion is discussed, with the conclusion that no acceleration of corrosion by radiation is to be expected in most cases of practical interest. (auth)

8738 CF-50-10-145
[Oak Ridge National Lab., Tenn.]
GAS GENERATION AND RADIATION DAMAGE IN THE HRE SOUP CIRCULATING PUMP. J. O. Bradfute. Oct. 26, 1950. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The quality of the O and H gases generated from the decomposition of water in the armature compartment of the soup pump of the HRE is discussed. The radiation energy released and the amount of decomposition of the pump oil are calculated. (F.S.)

8739 CF-51-4-57
Oak Ridge National Lab., Tenn.
RADIATION DAMAGE EXPERIMENTS FOR THE HOMOGENEOUS REACTOR. C. H. Secoy. Apr. 13, 1951. Decl. Feb. 14, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Tests of the effects of irradiation upon uranyl sulfate solutions were conducted using a highly enriched uranyl sulfate solution in light water contained in a type-347 stainless steel system. The solution is similar to that proposed as a fuel and heat transfer medium for the Homogeneous Reactor Experiment. Results of stability tests on the solution during irradiation are presented. Extensive studies of the corrosive effect of the solution showed excellent corrosion resistance of 18-8 stainless steels and of Ti at 100°C, but at 250°C corrosion of stainless steel accompanied by reduction of the uranyl ion resulted. Tests showed that pretreatment of the steel with either nitric or chromic acid provided a protective film of hydrous oxides, and prevented reduction of uranyl ion and loss of U from the solution after 10 months trial. Although such a protective film appeared adequate for the operation of a homogeneous reactor, 22 in-pile experiments were conducted to study the effects of radiation on the corrosion of the steel container and on the stability of the solution. Conditions for pretreatment for each of the 22 experiments, approximate temperature during irradiation, and duration of irradiation are given in table form. Results show that the corrosion behavior and solution stability is improved by the presence of pile irradiation. A possible explanation of this behavior is offered. Final tests of the behavior of circulating systems will be made in the Homogeneous Reactor Experiment. (C.H.)

8740 DC-51-6-9
Washington Univ., St. Louis.
THE EFFECT OF CHARGED PARTICLES ON THE PHYSICOMECHANICAL PROPERTIES OF MATERIALS. Monthly Report for the period May 1–June 1, 1951. W. P. Armstrong and C. J. Kippenhan. [nd]. Decl. Feb. 16, 1957. 9p. (GE-IY-1). $1.80(ph OTS); $1.80(mf OTS).

Report is made of measurements of the elastic modulus and electric resistivity of specimens containing 70% Be_2C–30% graphite. Bombardment with deuterons caused an initial sharp rise in resistivity followed by a period of fluctuating values, with probably a small over-all rising trend. Stress relieving of as received and degreased specimens by heating to 400°C gave evidence of a sharp initial rise in resistivity and decrease in elastic modulus followed by little or no change. Autoradiographs made from Al plates exposed at various beam currents showed that the deuteron beam becomes more diffuse as beam current increases. (C.H.)

8741 DC-51-7-10
Washington Univ., St. Louis. School of Engineering.
THE EFFECT OF CHARGED PARTICLES ON THE PHYSICO-MECHANICAL PROPERTIES OF MATERIALS. Monthly Report to Aircraft Gas Turbine Divisions General Electric Company, covering period June 1, 1951 to July

1, 1951. W. P. Armstrong and C. J. Kippenhan. [1951]. Decl. Feb. 16, 1957. 17p. (GE-IY-2). $3.30(ph OTS); $2.40(mf OTS).

Comparison of deuteron bombardments of stress-relieved (heat-treated) 70% Be_2C–30% graphite specimens with those not stress-relieved showed that the characteristic sudden large change in electric resistivity occurring at the start of bombardment is due almost entirely to a heat-treating effect. It was decided to run all future bombardments on uniformly stress-relieved (heat-treated) specimens. Autoradiographs of Al plates bombarded with the double dural foil beam window in place showed somewhat greater diffusion of the beam current density than with the windows out. Increased beam current resulted in increased diffusion of the beam density pattern. Additional work on stress relieving (by resistance heating in dry He atmosphere to 400°C) of as-received decreased specimens demonstrated the characteristic sharp rise of electrical resistivity and decrease in elastic modulus. Preliminary tests indicate that furnace stress relieving is comparable to resistance heating. (J.E.D.)

8742 HW-33029(Del.)

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

EXAMINATION OF RADIATION EXPOSED KEL-F PLASTIC COUPONS. C. L. Boyd. Sept. 10, 1954. Decl. with deletions Feb. 26, 1957. 5p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

8743 HW-35189

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

EXAMINATION OF IRRADIATED URANIUM–CHROMIUM ALLOYS SLUGS. W. K. Kratzer. Feb. 14, 1955. Decl. Feb. 26, 1957. 13p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The breaking strength of both the U–Cr alloy slugs and the normal U control slugs was reduced by exposure in the pile. The breaking strength of those U–Cr alloy pieces hot cracked or broken in the jackets before examination was consistently greater than the breaking strength of the normal U at all exposures. When examined, four of the five U–Cr slugs available from tube 4067-D, 834 MWD/T, were cracked or broken. Two of the alloy slugs from tube 4071-D, 687 MWD/T, were cracked longitudinally. None of the normal U control slugs cracked during pile exposure. One U–Cr alloy slug from tube 4071-D, 687 MWD/T, was visibly warped. None of the remaining alloy slugs or normal uranium slugs were found to be warped or distorted. The four U–Cr alloy slugs out of five (one of which was a rupture) that were cracked or broken as a result of exposure in the pile to 834 MWD/T indicate that these slugs are not as suitable for irradiation to high exposures as the normal U control slugs. The tendency of the alloy slugs to crack during exposure may have been aggravated by surface flaws present on the alloy cores before charging in the pile. (auth)

8744 LWS-22505

California Research and Development Co., Livermore, Calif.

LIFE EXPECTANCY OF ORGANIC GASKET MATERIAL AT THE TRANSITION-PRECESSOR POSITION WITHIN THE MARK I ACCELERATOR SHIELDING. S. G. Zizzo and W. E. Browning, Jr. Jan. 15, 1953. Decl. Feb. 28, 1957. 6p. (CRD-A17-41). $1.80(ph OTS); $1.80(mf OTS).

The life expectancy of elastomers as gaskets between the transition-precessor position are predicted suitable for 10^3 hours to 10^4 hours under the anticipated neutron flux conditions of 2.7×10^7 neutrons/cm^{2}sec. (auth)

8745 NAA-SR-73

North American Aviation, Inc., Downey, Calif.

STABILITY OF IRON TOWARD LITHIUM AT ELEVATED TEMPERATURES UNDER CYCLOTRON IRRADIATION. W. W. Parkinson, Jr. Aug. 27, 1952. Decl. Mar. 8, 1957. 47p. Contract AT-40-1-GEN-1064. $0.40(OTS).

Iron capsules containing Li were irradiated at temperatures of 625°C and 910°C, with the direct α beam of the Berkeley 60-in. cyclotron, to investigate radiation-induced corrosion. Chemical analysis of the Li charge showed no increase in corrosion at 625°C and indicated that there probably was little effect on corrosion at 910°C. Metallographic and x-ray studies carried out on the iron specimens indicated only slight structural changes, if any, attributable to irradiation at either temperature. (auth)

8746 NAA-SR-1079

North American Aviation, Inc., Downey, Calif.

PROTON IRRADIATION EFFECTS IN THORIUM. Charles J. Meechan. Dec. 15, 1954. Decl. Mar. 5, 1957. 16p. Contract AT-11-1-GEN-8. $0.30(OTS).

Iodide-processed Th foils were irradiated with 9-Mev protons at temperatures below -140°C. The recovery of electrical resistance upon annealing was studied in the range 0° to 75°C where tempering curves showed rapid changes taking place. Determinations of the activation energy associated with this process were made and the mean value obtained was 1.22 ev. Correlations of this result have been made with those found previously for Cu. From these comparisons, a tentative assignment of the motion of interstitial atoms in Th has been made for this process. In addition, some evidence has been found which illustrates the corrosive action that water has on Th at temperatures as low as 0°C. (auth)

8747 ORNL-1095

Oak Ridge National Lab., Tenn.

PHYSICS OF SOLIDS INSTITUTE QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1951. D. S. Billington and J. T. Howe, eds. Nov. 26, 1951. Decl. Mar. 5, 1957. 122p. Contract W-7405-eng-26. $19.80(ph OTS); $6.30(mf OTS).

The problems incident to making thermal-conductivity measurements in a reactor are discussed, and the apparatus is described. The data are shown for the change in resistivity of polycrystalline copper on cold working. A tensile calorimeter to measure the energy released in a sample cold-worked at the temperature of liquid N_2 is pictured, and its operation is described. Resistance measurements made before and after exposure of copper crystals to a cobalt γ source indicate that changes due to γ bombardment are negligible. Proton bombardment of annealed copper has produced a hardness increase of 20 DPH numbers on the Tukon tester. Variation of ionization energy with neutron bombardment in cuprous oxide is reported. Changes in electrical properties of Si due to fast-neutron bombardment have been measured. Evidence is given of hole traps in germanium resulting from fast-neutron bombardment. Inpile cantilever creep tests for stainless steel are reported. X-ray line-broadening studies of Cu–Be age-hardening alloy indicate a contraction of the lattice caused by an increase in the amount of Be. Chemical analyses of a pile-irradiated iron capsule containing liquid Li are given together with x-ray-diffraction measurements and photomicrographs. A description and pictures of the liquid-lithium loop are given. The γ-decay curves are presented for stainless steel, Zr, Ti, Al, and concretes. Radiation effects on Kovar and measurements of thermal-conductivity changes in uranyl nitrate hexahydrate are given. (C.H.)

SHIELDING

8748 CF-50-3-108
Oak Ridge National Lab., Tenn.
THE NEUTRON SPECTRUM EMERGING FROM AN IRON–WATER SHIELD. S. Podgor. Mar. 24, 1950. Decl. Feb. 18, 1957. 8p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

Calculations indicate that in an Fe-water shield of ⅔ Fe and ⅓ H_2O by volume, 1-Mev neutrons will tend to dominate in the emerging spectrum. This is on the assumption that scattering by H and inelastic scattering by Fe are the only important processes. (auth)

8749 CF-56-1-10
Oak Ridge National Lab., Tenn.
HEAT REMOVAL FROM IONIZATION CHAMBER SHIELD. N. F. Lansing. Jan. 3, 1956. 23p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

A discussion is given of the heat removal problems for the lead shield of the ionization chambers which are to be used as sensory devices for the ORR control system. Cooling passages embedded in the lead were suggested, and the design features of these passages are summarized. (B.J.H.)

8750 ORNL-424
Oak Ridge National Lab., Tenn.
APPROXIMATE ANALYSIS OF THE PENETRATION OF NEUTRONS THROUGH A THICK SHIELD COMPOSED OF NON-HYDROGENOUS MATERIALS. F. H. Murray. Dec. 14, 1949. Decl. Mar. 2, 1957. 30p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

URANIUM AND URANIUM COMPOUNDS

8751 A-1304
Metal Hydrides Inc., Beverly, Mass.
METAL HYDRIDES FINAL REPORT. P. P. Alexander. July 26, 1943. Decl. Feb. 12, 1957. 43p. $4.80(ph OTS); $2.70(mf OTS).

The commercial-scale production of U metal by Metal Hydrides Inc. is reviewed. The equipment used is shown. Included in appendices are data accumulated during the production of U metal. (B.J.H.)

8752 CT-2783
[Massachusetts Inst. of Tech., Cambridge.]
DEVELOPMENT OF PRECISION CASTING AND INVESTMENT MOLDING METHODS FOR TUBALLOY. Final Report. Oct. 12, 1944. Decl. Feb. 18, 1957. 33p. Contract W-7405-eng-175. $7.80(ph OTS); $3.30(mf OTS).

The "lost wax" technique using a silica bonded with ethyl silicate as the investment material was found to be the best way to cast U. The ways of preparing the wax patterns, molds for making these, methods of investing, drying, firing of the molds, melting and casting of the metal are given. (F.S.)

8753 K-905
Carbide and Carbon Chemicals Corp. K-25 Plant, Oak Ridge, Tenn.
CYLINDRICAL REACTOR DIMENSIONS OF THE WATER TAMPED ENRICHED URANYL FLUORIDE-WATER SYSTEM AS A FUNCTION OF CONCENTRATION. R. L. Macklin. May 9, 1952. Decl. Feb. 7, 1957. 17p. $0.30 (OTS).

The description of the cylindrical water-tamped, U^{235} and hydrogen core reactor experiments by empirical formulas has been extended. An analytic expression has been found for the functional dependence of the critical diameter of an infinite cylinder and the critical thickness of an infinite slab on moderation. Where it has been found possible to apply the formula of Schuske and Morfitt to the change of critical dimensions of water-tamped cylindrical reactors with identical high-assay U-solution core material, it has also been found possible to apply formulas of the type developed in the present report expressing the effect of change of solution concentration in the core. (auth)

8754 KAPL-1556
Knolls Atomic Power Lab., Schenectady, N. Y.
AN INVESTIGATION OF THE SINTERING OF URANIUM DIOXIDE. H. G. Sowman and G. L. Ploetz. Aug. 17, 1956. Decl. Feb. 26, 1957. 27p. Contract W-31-109-Eng-52. $0.35(OTS).

The sinterability of UO_2 obtained from various sources was investigated and found to vary considerably depending on the chemical process used for their preparation. Uranium dioxides prepared by the peroxide- and ammonia-precipitation methods were found to be superior to those prepared by the processes utilizing steam oxidation or burning. Some additives were found to improve the sintering characteristics. Additions of TiO_3 in amounts varying from 0.1 to 1.0 wt. % were especially effective. (auth)

8755 MCW-30
Mallinckrodt Chemical Works, St. Louis.
A GENERAL STUDY OF THE DISTRIBUTION OF IMPURITIES IN REMELTED URANIUM METAL AND RELATED SUBJECTS. W. A. Oppold. Apr. 1, 1946. Decl. Feb. 27, 1957. 14p. Contract W-7405-eng-29. $3.30 (ph OTS); $2.40(mf OTS).

Observation of pyrophoricity in rods produced from billets cast from remelted U metal, as well as need for basic knowledge of the distribution of impurities in remelted U, led to chemical and neutron-absorption analyses which indicated that Ag, Cr, Ni, B, Co, Fe, Mn, chloride, and silica are evenly distributed throughout the melt, whereas other impurities, including Mg, N, C, and H are much higher in the top portion of the melt, i.e., the last cast. Neutron absorption by the last metal in the crucible was very high. Billets from this last material were extremely pyrophoric, but this was attributed to microscopic porosity rather than chemical impurity. (G.Y.)

8756 NAA-SR-255
North American Aviation, Inc., Downey, Calif.
HIGH TEMPERATURE DIFFUSION OF INDIVIDUAL FISSION ELEMENTS FROM URANIUM CARBIDE IMPREGNATED GRAPHITE. L. B. Doyle. Sept. 11, 1953. Decl. Mar. 4, 1957. 42p. Contract AT-11-1-GEN-8. $0.40 (OTS).

The diffusion of major fission elements from previously neutron-irradiated, U carbide-impregnated graphite has been studied at 1500, 1700, and 1900°C in vacuo. Data were taken at selected time intervals up to 24 hr for each element at each temperature. Cs and Sr were found to diffuse rapidly at 1500°C. Ba, Te, and I diffused rapidly at 1900°C. The rare earths diffused slowly even at 1900°C, except for Pr, which diffused several times as rapidly as Ce and Y. Ru, Mo, and Zr were largely retained in the graphite at 1900°C. Diffusion curves for rapidly diffusion elements are plotted and interpreted on the basis of diffusion theory. Ba and Pr (and, by inference, the other rare earths) were found to have diffusion rates dependent on graphite specimen size; but I, Te, and perhaps Sr diffused at rates independent of specimen size. Escape into the pores of the graphite from interstital positions in the graphite crystallites is postulated as the rate-limiting process whereever diffusion is independent of specimen size. (auth)

8757 NAA-SR-Memo-181
North American Aviation, Inc., [Downey, Calif.]
THERMAL STRESSES IN A LONG COMPOSITE CIRCULAR URANIUM ROD. J. Brandstatter. Jan. 28, 1952. Decl. Feb. 27, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

This brief memo concerns the thermal stresses in an infinitely long uranium rod of circular cylindrical geometry. The media of the rod consist of α and β phase uranium, with the β-phase portion occupying the central portion of the rod. The α-phase uranium composes the outer core of the cylinder. It is assumed that the β-phase extends from the center of the rod to a radius of 0.9525 centimeter. The α-phase extends from 0.9525 to 1.270 centimeters; i.e., the radius of the rod. The temperature distribution in the rod is assumed to have the following form, $T(r) = T_0 - (\Delta T)(r/b)^2$, where T_0 is the temperature at the center of the rod, and ΔT is the temperature difference that exists between the center of the rod and the external radius, b, of the rod. The thermal stresses are computed from the formulas by Gatewood (Phil. Mag., J. of Science 32, 282(1941)). (auth)

8758 TID-5183
Los Alamos Scientific Lab., N. Mex.
SEARCH FOR DOPPLER EFFECT IN THE HEATING OF U^{235}. David B. Hall and Jane H. Hall. Dec. 9, 1948. Decl. Feb. 7, 1957. 22p. Contract W-7405-eng-36. $0.35 (OTS).

Several experiments were undertaken in order to detect temperature effects on the Doppler broadening of fission and absorption resonance lines in U^{235}. It is pointed out that rapid temperature increases in a reactor fuel rod could cause a nuclear explosion if an increase in reactivity occurred concurrent with high local heating of the fuel rods. The effect was studied by heating a rod of U^{235} and an Al rod, noting differences in the reactivity of a Pu reactor. In all cases, the effect was not found. (K.S.)